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ABSTRACT

This technical specialty course is one of thirty-five courses designed for use by two-year postsecondary institutions in five nuclear technician curriculum areas: (1) radiation protection technician, (2) nuclear instrumentation and control technician, (3) nuclear materials processing technician, (4) nuclear quality-assurance/quality-control technician, and (5) nuclear reactor (plant) operator trainee. This course, Introduction to Nuclear Systems, contains five modules: Basic Nuclear Reactor Components, Pressurized Water Reactors, Boiling Water Reactors, Primary Cooling Systems, Nuclear Instrumentation, and Nuclear Fuel Cycle. Each module follows a typical format including the following sections: introduction, module prerequisites, objectives, subject matter, exercises, materials, procedures, data tables, reference materials, and tests. (A master index and suggested program planning guides are also available--see CE 026 856 and CE 026 639-642.) (LRA)

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ED192148

Nuclear Technology Series

COURSE 5
INTRODUCTION TO
NUCLEAR SYSTEMS

TECHNICAL EDUCATION RESEARCH CENTER - SW

4201 LAKE SHORE DRIVE - SUITE 111

WACO, TEXAS 76710

CE 026 779

U S DEPARTMENT OF HEALTH, EDUCATION & WELFARE
NATIONAL INSTITUTE OF EDUCATION

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PREFACE

ABOUT NUCLEAR TECHNOLOGY MODULES

The modules - developed by TERC-SW for use in two-year postsecondary technical institutions to prepare technicians for employment - are useful in industry for updating employees in company-sponsored training, and as laboratory texts in engineering schools. The principles, techniques and skills taught in the modules, based on tasks that nuclear technicians perform, were obtained from a nationwide advisory committee of employers of nuclear technicians. Each module was written by a working technical expert and tested with students.

A module contains the following elements:

Introduction, which identifies the topic and often includes a rationale for studying the material.

Module Prerequisites, which identify the material a student should be familiar with before studying the module.

Objectives, which clearly identify what the student is expected to know for satisfactory module completion. The objectives, stated in terms of action-oriented behaviors, include such action words as operate, measure, calculate, identify and define, rather than words with many interpretations, such as know, understand, learn and appreciate.

Subject Matter, which presents the background theory and techniques supportive to the objectives of the module. Subject matter is written with the technical student in mind.

Exercises, which provide practical problems to which the student can apply this new knowledge.

Materials, which identify the equipment required to complete the laboratory procedure.

Procedures, which is the experiment section, or "hands-on" portion of the module (including step-by-step instruction) designed to reinforce student learning.

Data Tables, which are included in most modules for the first year (or basic) courses to help the student learn how to collect and organize data.

Reference Materials, which are included as suggestions for supplementary reading/viewing for the student.

Test, which measures the student's achievement of the prestated objectives.

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The Nuclear Technology Series will consist of five curricula when completed:

Nuclear Reactor (Plant) Operator Trainee.
Radiation Protection Technician.
Nuclear Materials Processing Technician.
Nuclear Quality-Assurance/Quality-Control Technician.
Nuclear Instrumentation and Control Technician.

The completed Nuclear Reactor (Plant) Operator Trainee curriculum consists of the following sixteen technical specialty courses made up of 114 instructional modules:

Radiation Physics.
Radiation Protection I.
Principles of Process Instrumentation.
Heat Transfer and Fluid Flow.
Introduction to Nuclear Systems.
Instrumentation and Control of Reactors and Plant Systems.
Reactor Operations.
Reactor Safety.
Reactor Auxiliary Systems.
Power Plant Systems.
Radiation Detection and Measurement.
Reactor Physics.
Power Plant Chemistry.
Introduction to Quality Assurance/Quality Control.
Metallurgy and Metals Properties.
Mechanical Component Characteristics and Specifications.

Availability: Instructional modules are available individually or by courses from:

Technical Education Research Center - Southwest
4201 Lake Shore Dr., Suite 111
Waco, Texas 76710
Telephone (817) 772-8756

Printed:

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Module 5-2 Pressurized Water Reactors (PWR)

Module 5-3 Boiling Water Reactors (BWR)

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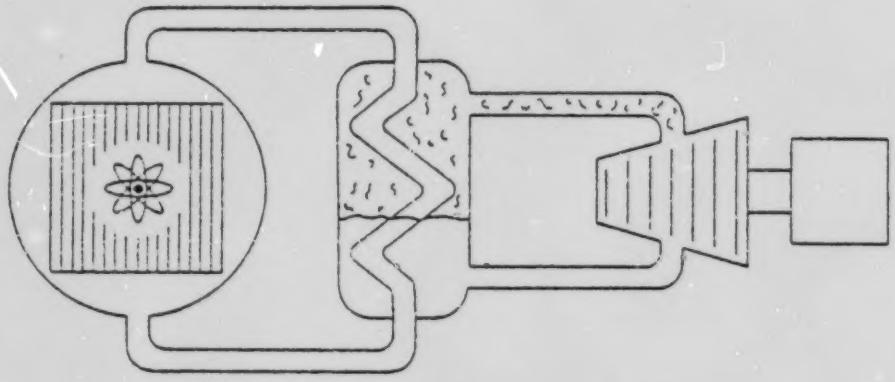
Module 5-5 Nuclear Instrumentation

Module 5-6 Nuclear Fuel Cycle

NUCLEAR TECHNOLOGY

MODULE 5-1 BASIC NUCLEAR REACTOR COMPONENTS. -

GEORGE J. STOLL, P.E.



INTRODUCTION TO NUCLEAR SYSTEMS

COURSE 5

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST

4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 5 INTRODUCTION TO NUCLEAR SYSTEMS

MODULE 5-1 Basic Nuclear Reactor Components

MODULE 5-2 Pressurized Water Reactors (PWR)

MODULE 5-3 Boiling Water Reactors (BWR)

MODULE 5-4 Primary Cooling Systems

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MODULE 5-6 Nuclear Fuel Cycle

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In recent years, there has been a tremendous increase in the demand for electrical power by individuals and industrial concerns. The major sources of this power have been steam powered electrical generation plants, in which steam drives turbines, which then drive generators, to produce electricity (Figure 5-1-1). In a conventional pow-

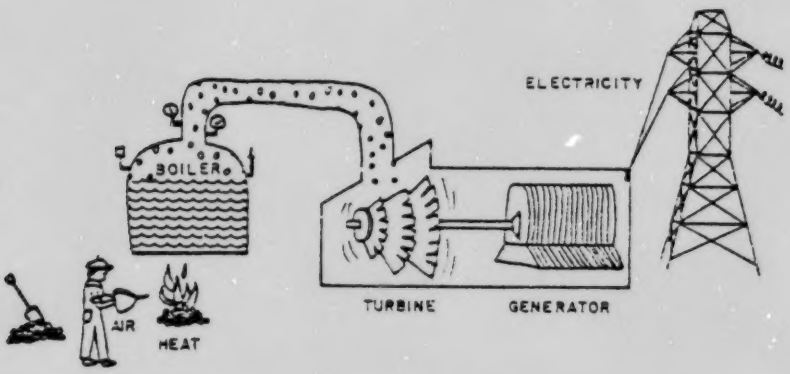

Figure 5-1-1. Fossil-fueled steam/electric power generation.

er plant, oil, coal, natural gas or other combustible fuel is burned to produce the steam. In a nuclear reactor power plant (Figure 5-1-2), on the other hand, steam is produced from energy released when uranium atoms split (fission).

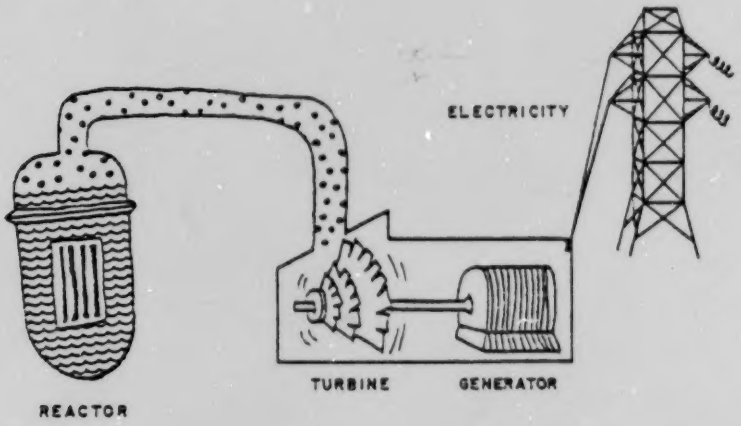

Figure 5-1-2. Nuclear-fueled steam-electric power generation.

MODULE PREREQUISITES

The student should have completed Course 4, "Heat Transfer and Fluid Flow," and its prerequisites.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. List six components common to all light-water moderated and cooled reactors.
2. List four design objectives of light-water reactors.

3. Briefly define or describe a BWR and a PWR.
4. Sketch a PWR and a BWR plant; show major components; describe coolant flow.
5. Sketch cross-sectional views of a PWR and a BWR vessel; show major components, penetrations, and coolant flow direction.
6. Describe principles of operation of three control-rod drive systems.
7. Sketch typical PWR and BWR control rods.
8. Sketch a PWR and a BWR fuel assembly.
9. Define:
   - Fuel.
   - Moderator.
   - Coolant.
   - Steam generator.

LIGHT-WATER REACTORS

The end result of a nuclear chain reaction process, in any reactor, is the production of heat and radiation. The desired product of a power reactor is heat. Efficient utilization of this heat in the production of electricity by a steam turbine generator is the greatest concern in the nuclear steam cycle.

Reactors that have attained the greatest development and technology (in the United States) are light-water moderated and cooled plants, fueled by uranium (that has been slightly enriched in the isotope ^{235}U) in stainless steel or zirconium cladding. Such reactors consist of the following systems:

Fuel, containing UO_2 which fissions after absorbing "slow" neutrons.

Moderator, which reduces energy of neutrons to a level at which they can be absorbed by the fuel.

Control devices, which vary the number of neutrons available (in a core) for absorption by the fuel.

Coolant system for heat removal.

Instrumentation, plant control and performance devices.

Containment vessel, with structures and shielding materials.

Design objectives include:

- Economical fuel cycle.
- High thermal efficiency.
- Safety and reliability.
- Neutron economy.

Although there are many types of reactor designs in the United States, light-water reactors are the only ones currently being designed and fabricated for utility use. These may be classified broadly in two types: pressurized water reactor (PWR) and boiling water reactor (BWR).

A PWR is a solid-fuel, thermal reactor that uses ordinary water as a moderator and coolant. It contains an intermediate coolant loop between the reactor and turbine to avoid contamination of the steam system and the atmosphere. Water, in the primary loop, is maintained at a pressure in the 2000-pound psi range to keep system pressure above the saturation pressure associated with the system operating temperature.

The primary system of a PWR includes a reactor vessel, steam generators, pressurizer, main coolant piping, valves and pumps (Figure 5-1-3).

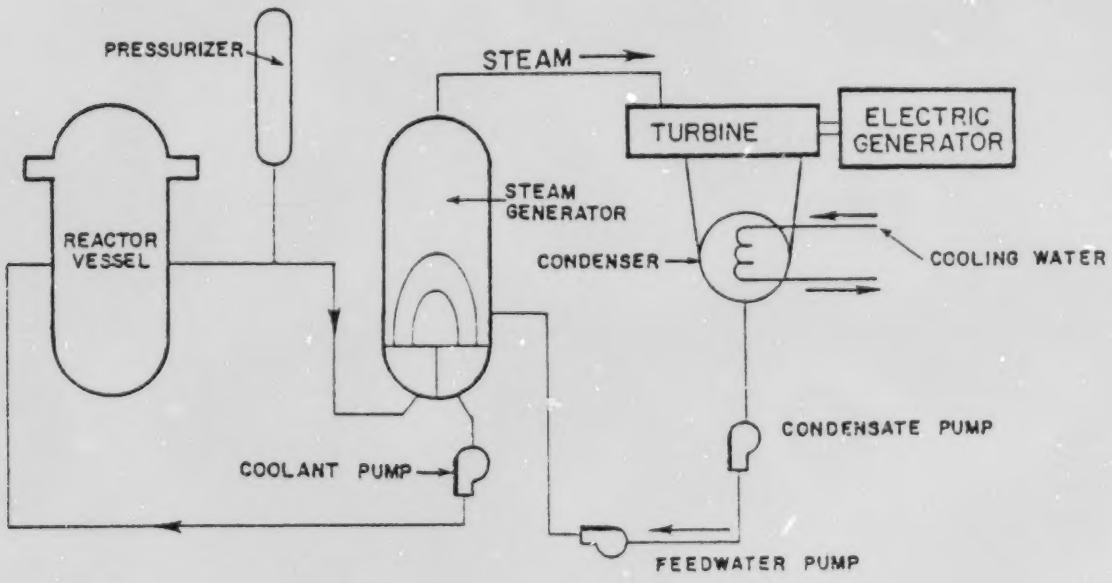

Figure 5-1-3. Typical pressurized water reactor (PWR) system.

The steam generator is actually a heat exchanger in which heat from the pressurized water in the primary loop is transferred to water in the secondary loop (maintained at a lower pressure). Steam is generated (hence the name) in the secondary loop because the actual temperature is above the boiling point corresponding to that pressure.

Disadvantages of this type of reactor include 1) its necessity for an intermediate heat exchanger and the temperature drop through that heat exchanger, 2) the shielding required around the primary loop equipment and 3) the need for heavier components in the primary loop and reactor vessel.

A BWR is a direct cycle system (Figure 5-1-4). Water boild in the reactor core, producing saturated steam, which flows directly to the turbine. This is a relatively simple

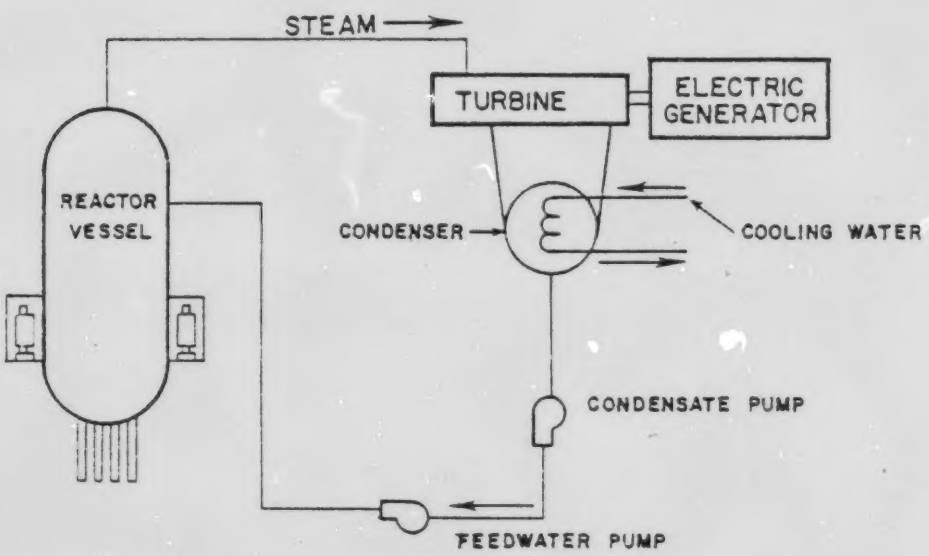

Figure 5-1-4. Typical boiling water reactor (BWR) system.

system with advantages in performance and cost. It does not have an intermediate heat exchanger for the transfer of heat from the coolant to another medium. The steam temperature at the turbine throttle is nearly the same as a reactor outlet temperature and, because the control rods are installed in the bottom of the reactor, refueling is less difficult. The major problem, however, is the carry-over of radioactive substances to the steam equipment which can make system maintenance difficult.

An important fact about both the PWR and BWR systems is that the primary coolant circulates within a closed circuit and does not contact its original heat sink (river, lake or ocean). In commercial nuclear plants, outside water is used for condenser cooling. It is important to note that none of this water flows through the reactor. Its only purpose is to carry unusable heat away from the system.

In the reactor cooling process, the reactor coolant becomes radioactive. One source of this radioactivity is leakage from the fuel element containment; however, only 1/1000th of one percent of the fission product inventory of the plant is involved, and it is principally gaseous in nature. Another source of radioactivity is plant activation products. Most have short half-lives. These products are activated in the reactor structure and picked up by the coolant through corrosion or erosion.

Water in the primary coolant system is maintained and purified to limit the amount of radioactivity in the system. This limiting is accomplished by drawing off a portion of the primary coolant flow and passing it through purification equipment, with the subsequent return to the system.

Both PWR's and BWR's require extensive shielding and

containment buildings to contain releases and debris in the event of a major rupture or failure of the coolant circuit. These structures must be able to withstand the relatively large internal pressures that could occur with releases of radioactive gases and steam from the reactor.

Cost and fabrication procedures favor cylindrical buildings with hemispherical heads and a flat foundation slab for PWR's (Figure 5-1-5). These buildings are reinforced

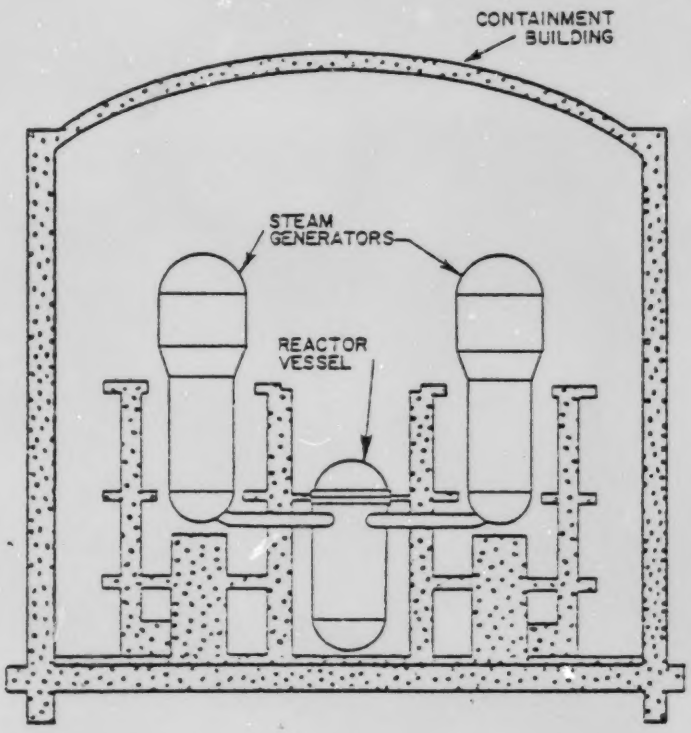

Figure 5-1-5. Typical PWR containment building.

with high tensile strength cables in tubes placed circumferentially around the building with further longitudinal

tubes and cables terminating in the head ring. The structure is then prestressed by tensioning the cables after installation.

A typical BWR containment building is quite similar (Figure 5-1-6). An upperpool and a suppression pool are

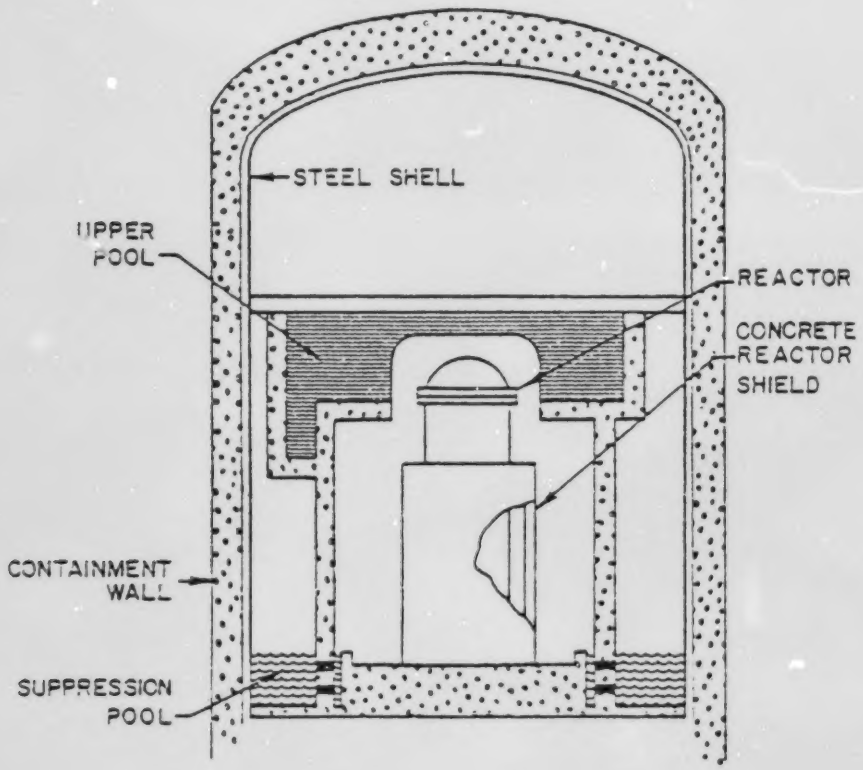

Figure 5-1-6. Typical BWR containment building.

used to condense steam during a loss-of-coolant accident (LOCA), thereby reducing pressure inside the containment building. (Construction of these buildings is described in greater detail in Module 9-1, "Containment and Ventilation of Reactors.")

REACTOR VESSEL

Primary functions of a reactor vessel are core containment, primary coolant flow direction and guiding of control rods.

This vessel is fabricated from bent plate or cylindrical forgings, ranging from approximately 40 to 75 feet in overall height and 15 to 20 feet in inside diameter, according to the manufacturer and type of vessel involved. It should be noted that BWR vessels (Figure 5-1-7) must be

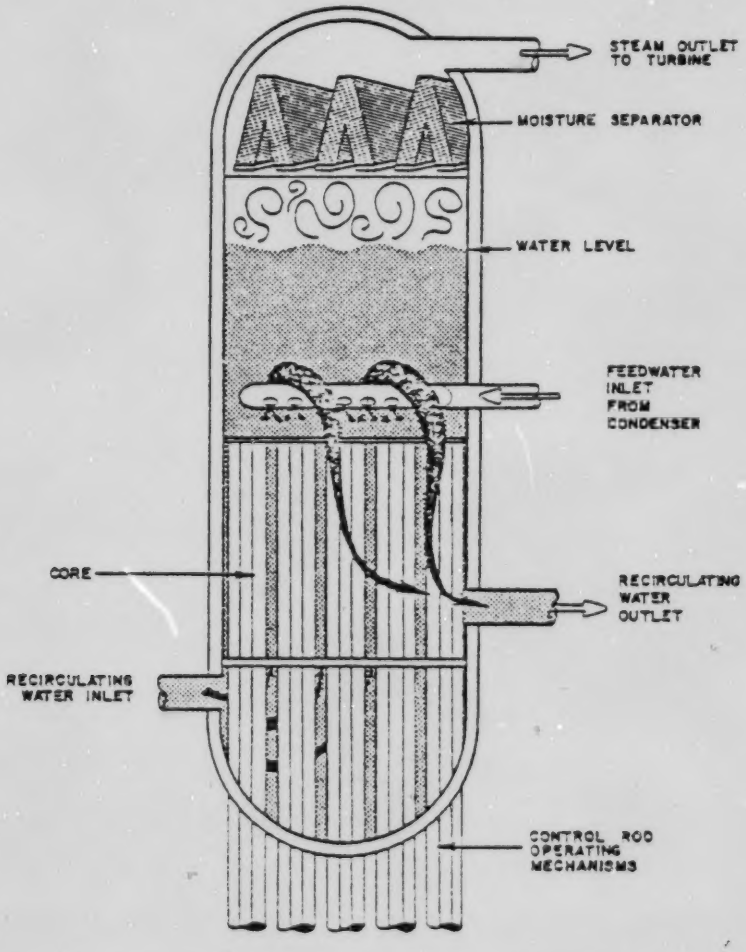

Figure 5-1-7. BWR vessel components (simplified).

larger than PWR vessels (Figure 5-1-8) because of their

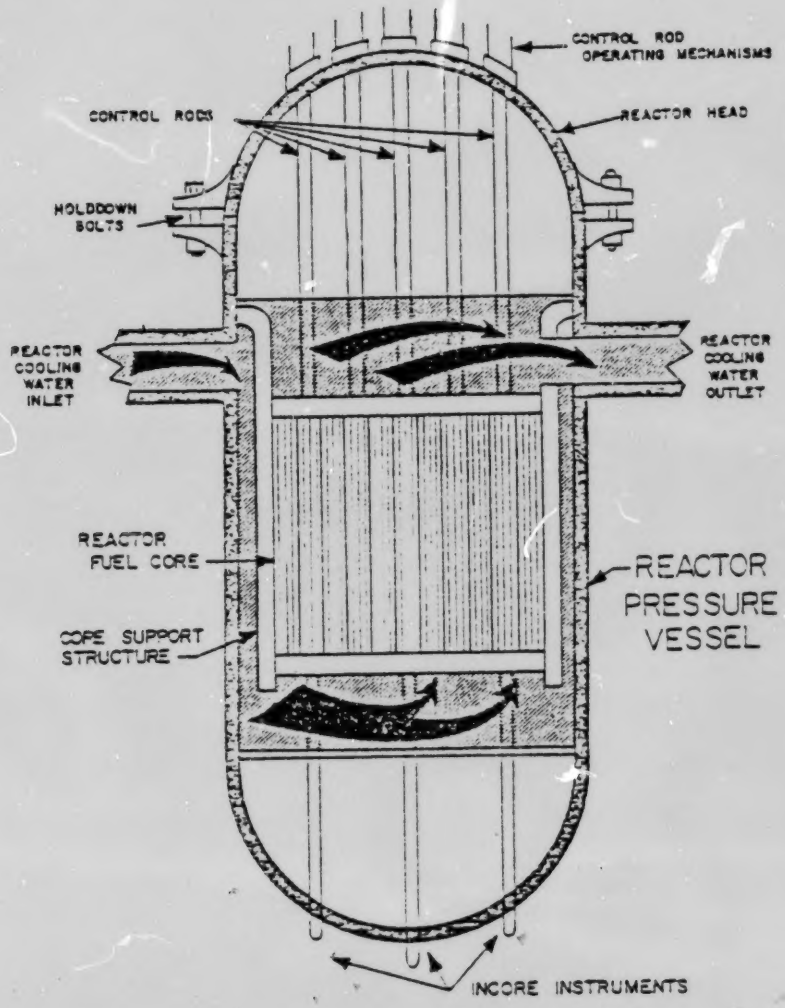

Figure 5-1-8. PWR vessel components (simplified).

larger core and steam separating equipment (in the top of the vessel). (Vessel components are discussed in greater

detail in later modules in this Course.)

All major vessel penetrations are above the top of the core so that, in the event of a pipe rupture, fuel will remain covered by water.

Reactor internals are made of stainless steel, Inconel (a corrosion-resistant alloy of nickel, chromium and iron) or one of several zirconium alloys. The vessel itself is a low-alloy manganese-molybdenum steel with internal surfaces clad with stainless steel. It is designed to Class I of Section III of the ASME Code, plus Seismic I Code, and needs a code stamp.

CONTROL-ROD DRIVE SYSTEMS

If the power level of a reactor must change to meet system requirements, it is necessary to have a control mechanism. In the case of power reactors, this is accomplished by control rods in combination with soluble poison (boric acid, which is a material with the ability to absorb neutrons) dissolved in coolant water.

PWR SYSTEMS

Reactor power level is controlled by in and out movement of control rods within the core, combined with varying the chemical shim (quantity of boric acid) in the primary coolant.

The combination of neutron-absorbing qualities of the rods and boric acid makes it possible to change the level

of power while maintaining temperature and pressure parameters.

Two systems of rod-drive mechanisms are used in PWR systems:

Magnetic jack system (Figure 5-1-9). This mechanism

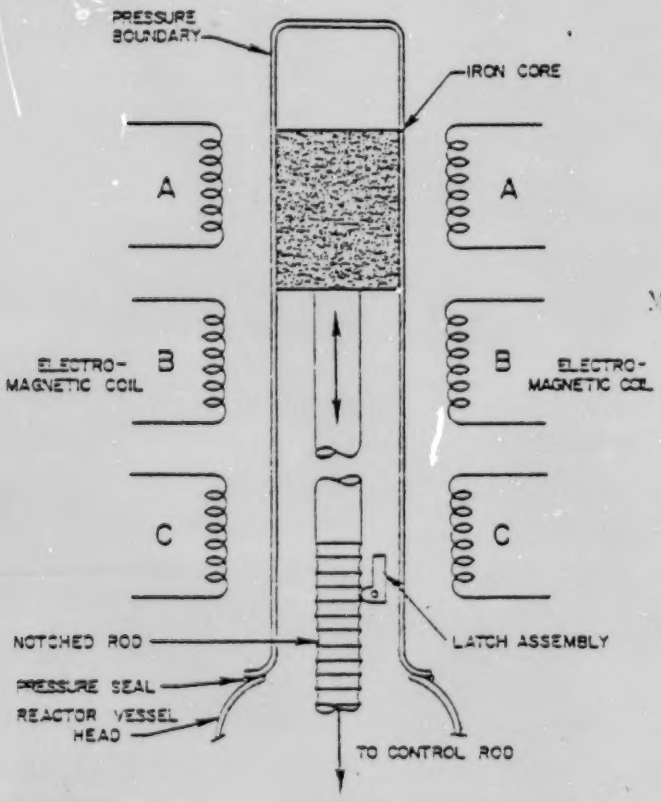

Figure 5-1-9.
Magnetic jack control-rod drive system.

is fully enclosed (zero leakage) and consists of a set of magnets (A, B and C) mounted outside the pres-

sure tube. The notched rod is jacked up and down in short increments by sequential stepping of the magnet set. When the magnets are de-energized, the rods fall into a "scram" position.

Roller nut type (Figure 5-1-10). This type uses an

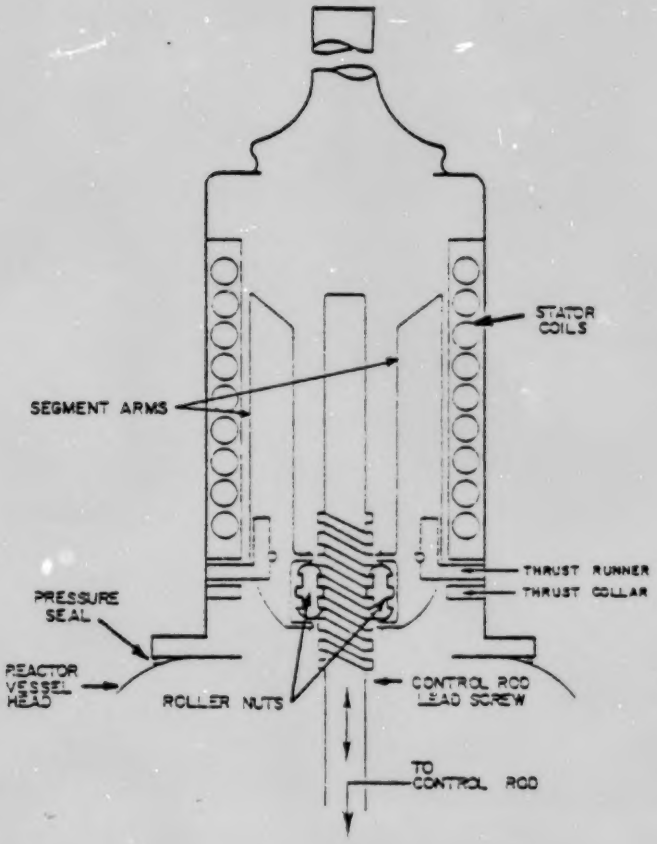

Figure 5-1-10. Roller nut control-rod drive assembly.

electromechanical device, which converts rotary motion (supplied by a magnetic field) into up and down

motion of the lead screw. For rapid insertion, the magnetic field is de-energized, and the rod falls into the core (pulled by gravity).

Control rods, which attach to either of the above drive systems, consist of cylindrical rods attached to spiders. They are situated inside control-rod guide tubes in the fuel assemblies (Figure 5-1-11). Each cluster of from 5 to 20 rods is attached to a separate drive assembly.

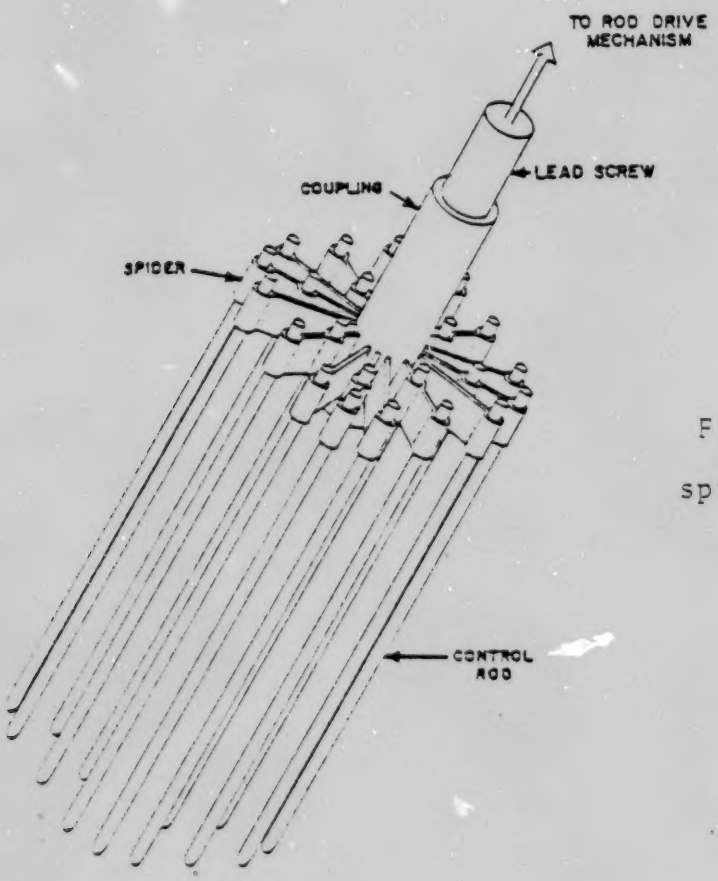

Figure 5-1-11. Control-rod spider assembly.

BWR SYSTEMS

In a BWR system, control rods are energized hydraulically with rod motion functionally opposite to methods normally used in PWR systems. Control rods enter the vessel from the bottom of the reactor. Inward and outward rod motion is controlled by a series of valves and hydraulic fluid. A reactor "scram" is accomplished by the action of overpressure stored in an accumulator, which is independent of outside power.

The shape of the control rod is usually a cruciform (Figure 5-1-12), containing compacted boron carbide powder

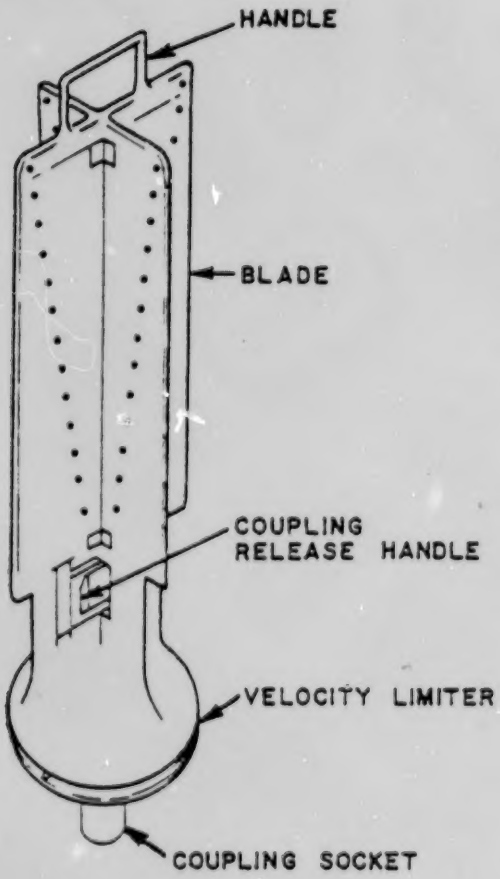

Figure 5-1-12.
BWR
control rod.

in a stainless steel cover with handles and a parachute-shaped bottom casting. Vanes of the cruciform have holes to permit water to flow through the assembly to cool the rod and to equalize water pressure, thereby reducing dynamic resonance vibration.

FUEL ASSEMBLIES

The reactor core is the furnace of the nuclear steam supply system (NSSS), with heat being generated in the fuel assemblies. In a PWR system, pressurized coolant water transfers heat generated in the fuel to the secondary water in the steam generators (producing steam for turbogenerators). In a BWR system, fuel assembly transfers heat to the primary coolant, producing steam for the turbogenerators. The water in both systems also acts as a moderator to slow down neutrons to the thermal range, permitting fission to take place.

PWR SYSTEMS

Fuel assemblies are arranged in the core on a square lattice (Figure 5-1-13). Individual fuel assemblies are made up of fuel rods, control-rod guide tubes and instrument tubes. The basic element of the core is the fuel pellet, which is uranium dioxide (UO_2) powder formed into a cylinder (by cold pressing) and sintered for density. These pellets are enclosed in a Zircaloy tube, pressurized with

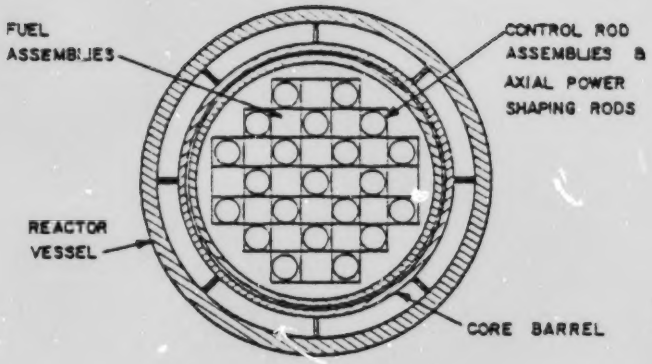

Figure 5-1-13.
PWR fuel assembly
arrangement.

helium and welded at each end (Figure 5-1-14). The rods

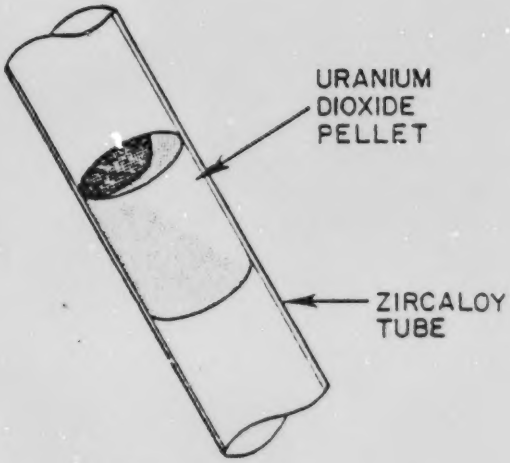

Figure 5-1-14.
Uranium dioxide
pellet enclosed in
Zircaloy tube.

are then assembled and bound together with other components to form an "assembly" (Figure 5-1-15). All fuel assemblies are mechanically identical and are installed vertically on the core support plate in the reactor vessel.

After installation of all assemblies, the upper sup-

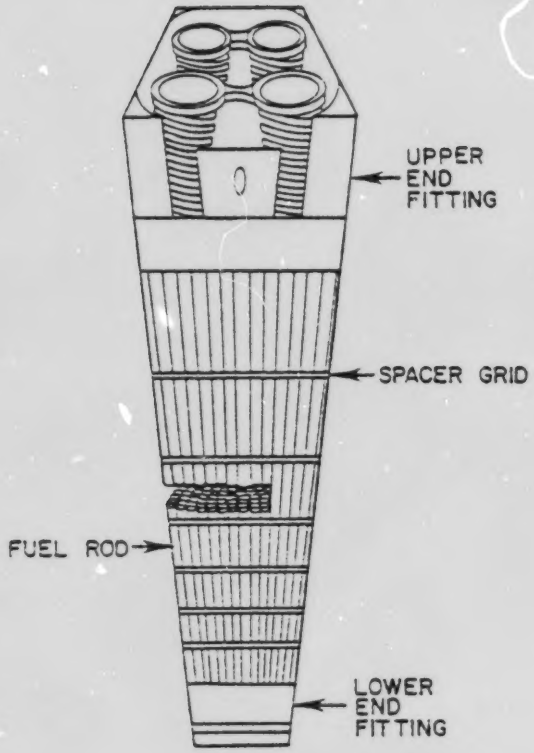

Figure 5-1-15.
Typical PWR fuel
assembly.

port structure is installed. The number of assemblies, rods in an assembly and fuel length vary by vendors (Table 5-1-1).

The core for an initial fuel load contains pellets with enrichments of ^{235}U, ranging from 1.6 wt% to 2.2 wt% (% of total of all uranium), depending on fuel cycle requirements. Reload requirements range in enrichment from 2.4% to 2.8%.

TABLE 5-1-1
BASIC PARAMETERS - PWR SYSTEMS

	NSSS VENDORS		
	Westinghouse Std. 1300 MWe Plant	Combustion Engineering System 80	Babcock and Wilcox B-205
Core heat output, Mwt	3800	3800	3800
System oper. press., nominal, psia	2250	2235	2250
Fuel assemblies			
No. fuel assemblies	193	241	205
Rod pitch, in.	0.496	0.5063	0.501
Fuel pellet diam., in.	0.3225	0.325	0.324
Fuel pellet length, in.	0.530	0.390	0.375
Clad material	Zirc 4	Zirc 4	Zirc 4
Reactor core structure core diam., in. (equivalent)	132.7	157	-
Coolant piping			
No. loops	4	2	2
Hot leg ID, in.	29	42	32
Cold leg ID, in.	27.5	30	38
Coolant pumps (each)			
Type	Vertical stainless steel centrifugal	Vertical stainless steel centrifugal	Vertical carbon steel centrifugal
Design capacity, gpm	97,600	111,400	108,500
Design head, ft.	343	363	376
Coolant pump motors			
No. per system	4	4	4
Horsepower	8000	9000	9000
Vessel			
Forging material	Carbon steel	Carbon steel	Carbon steel
Clad material	Stainless steel	Stainless steel	Stainless steel
Nominal inlet temp., °F	559.8	564.5	569
Nominal outlet temp., °F	623.8	621.2	626
Design pressure, psia	2500	2485	2500
Design temp., °F	650	650	670
Steam generator (each)			
Type	Vert. U-tube	Vert. U-tube	Straight tube

BWR SYSTEMS

As with PWR fuel assembly, BWR fuel assembly starts with UO_2 pellets in Zircaloy tubing.

These rods are then assembled into fuel bundles of 64 fuel rods in an 8 x 8 array supported by an upper and a lower tie plate (Figure 5-1-16).

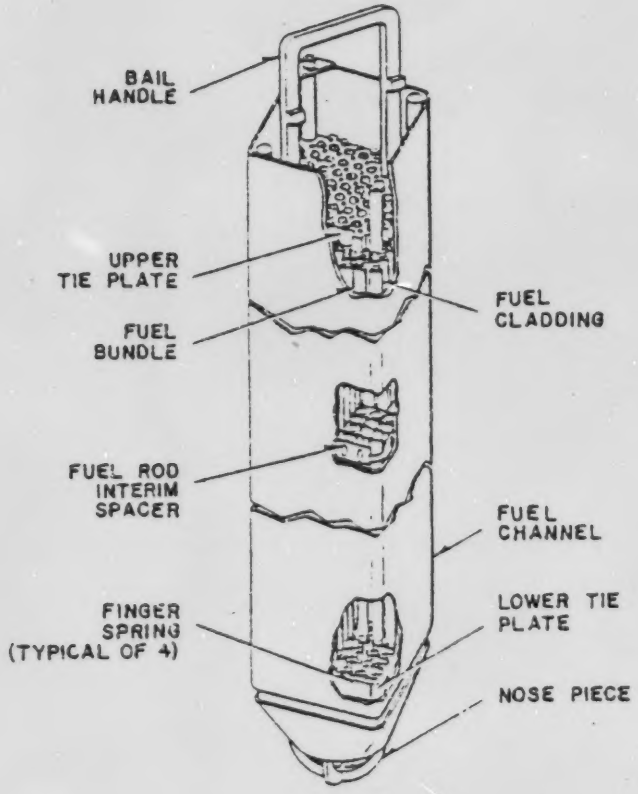

Figure 5-1-16. Typical BWR fuel assembly.

In summary, there are two broad types of nuclear reactor power plants. Basic concepts and components are the same for PWR and BWR plants, but there are some important distinctions.

The most important distinction is that, in a BWR, steam is produced in the reactor coolant system, while in a PWR, reactor coolant is separated from the turbines by a steam generator (in which the steam is produced). The methods may vary, but the results are the same: electricity from the energy of nuclear fission.

EXERCISES

None.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

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General Electric Co., Nuclear Energy Division. General Description of a Boiling Water Reactor. San Jose, CA: General Electric Co., Nuclear Energy Division, May, 1974.

Lish, Kenneth C. Nuclear Power Plant Systems and Equipment. New York: Industrial Press, 1972.

1. List six components common to all light-water moderated and cooled reactors:
   (1)
   (2)
   (3)
   (4)
   (5)
   (6)
2. List four design objectives of light-water reactors:
   (1)
   (2)
   (3)
   (4)
3. Write an appropriate definition for PWR and BWR plants:

4. Sketch a PWR and a BWR plant; show major components, describe coolant flow:

5. Sketch cross-sectional views of a PWR and a BWR pressure vessel; show major components, penetrations and direction of coolant flow:

6. Briefly describe principles of operation of these control-rod drive systems:

7. Sketch typical PWR and BWR control rods:

8. Draw a sketch of typical PWR and BWR fuel assemblies:

9. Define:

Fuel.

Moderator.

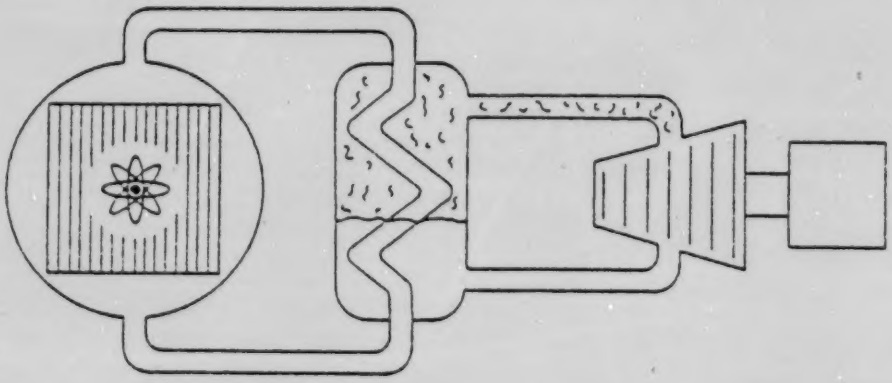
Coolant.

Steam generator.

NUCLEAR TECHNOLOGY

MODULE 5-2 PRESSURIZED WATER REACTORS (PWR)

GEORGE J. STOLL, P.E.



INTRODUCTION TO NUCLEAR SYSTEMS

COURSE 5

TERC-SW

34

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST

4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE 5 · INTRODUCTION TO NUCLEAR SYSTEMS

MODULE 5-1 Basic Nuclear Reactor Components

MODULE 5-2 Pressurized Water Reactors (PWR)

MODULE 5-3 Boiling Water Reactors (BWR)

MODULE 5-4 Primary Cooling Systems

MODULE 5-5 Nuclear Instrumentation

MODULE 5-6 Nuclear Fuel Cycle

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INTRODUCTION

Three of the present four manufacturers of nuclear reactors for power plant service in the United States fabricate pressurized water reactor (PWR) systems. These three manufacturers have standardized their products in the range of 1200 MWe output with similar configurations, except for steam generators. Two companies use saturated steam U-tube designs; the other uses a once-through design that produces superheated steam.

This module describes principal components and operating characteristics of typical PWR's.

MODULE PREREQUISITES

The student should have completed Module 5-1, "Basic Nuclear Reactor Components," and its prerequisites.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Sketch a PWR vessel, showing core, core support structure, reactor head, control rods, reactor

vessel and reactor coolant flowpath.

2. Describe the design and operating characteristics of a typical reactor coolant pump.
3. Sketch and explain operation of a typical pressurizer.
4. Sketch two types of steam generators; using sketch to explain operation of each.
5. Sketch two types of PWR control-rod drive mechanisms; explain operation of each.
6. Explain why boron is added to PWR coolant.
7. Sketch and explain operation of a typical PWR secondary loop.

REACTOR VESSEL

The reactor vessel is the heart of the reactor coolant system. This vessel contains the core and it provides a coolant path. Usually, it consists of a spherical, dished-head bottom welded to a cylindrical main shell. The vessel is sealed at the top by a closure head, which is bolted to the top flange of the vessel (Figure 5-2-1).

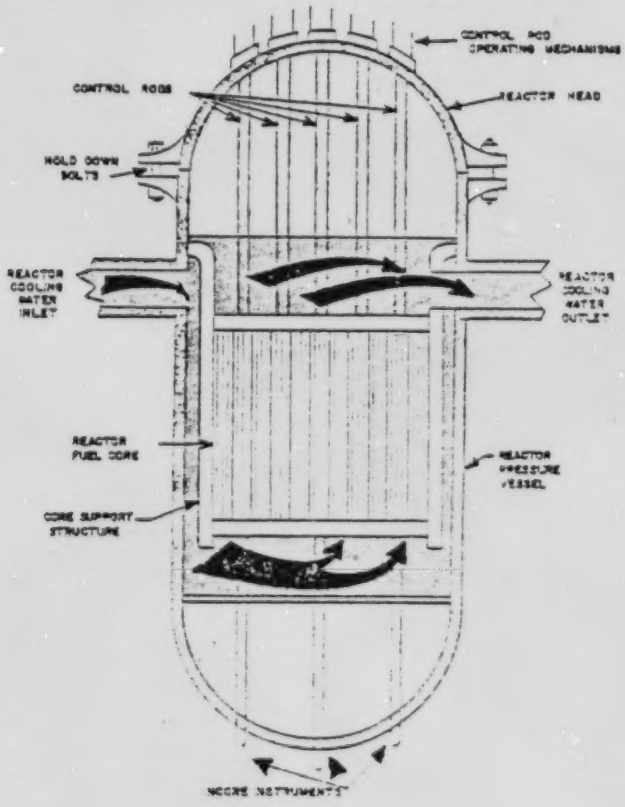

Figure 5-2-1. Typical reactor pressure vessel.

The vessel body is constructed of formed steel plates,

welded longitudinally to form cylindrical sections, or of integral steel forgings in the form of cylinders. Usually a low-alloy carbon steel is used, with the inside surface clad with stainless steel to minimize corrosion.

The vessel contains the core, core supports, the control-rod guide tubes, incore-instrument guide tubes, control-rod clusters, core peripherals and, sometimes, a thermal shield.

Reactor vessel penetrations usually consist of inlet and outlet nozzles, control-rod drive flanges and core-flood nozzles. These major penetrations are situated above the top of the core so that fuel assemblies remain covered by water, even if a pipe ruptures. The bottom header is usually penetrated by only incore-instrumentation guide tubes.

The vessel is supported within the reactor building by pads, or other devices, mounted beneath the coolant inlet nozzles and attached to concrete, or steel, supports built into the containment.

REACTOR INTERNALS

The reactor internals support the core, maintain alignment of the fuel assemblies, limit movement of these assemblies and maintain alignment of control-rod guide tubes between fuel assemblies and control-rod drives. Also, the internals direct the flow of reactor coolant, provide gamma and neutron shielding, provide guides for incore instrumentation between the lower head of the reactor vessel and fuel assemblies. Further, they support surveillance specimen assemblies (in the annulus between the core-support assembly and reactor vessel wall).

Reactor internals are designed to withstand stresses resulting from startup, from steady-state operations with two, three or four reactor coolant pumps running, and from shutdown conditions. Loss of pumping power does not damage them. Stresses, determined by load conditions in various modes of operation, include hydraulic forces, differential thermal expansions, component weights and seismic (loads as specified in the ASME Boiler and Pressure Vessel Code).

The internals are designed to resist the effects of earthquake disturbances. Lateral deflection and torsional rotation, limited to prevent excessive movements resulting from seismic disturbances, prevent interference with control-rod assemblies. If normal supports fail, core drop is limited so that control-rod assemblies do not disengage from their fuel-assembly guide tubes. All internal components can be removed from the reactor vessel to allow inspection of components and internal surfaces of the vessel (Figure 5-2-2).

A core support shield which supports the core barrel, lower grid assembly and reactor core, is made up of a ring forging (or welded plates). It includes the assembly support flange, outlet nozzle forgings, shell courses and lower support flange section, which supports the lower grid assembly and core basket on its internal ledge.

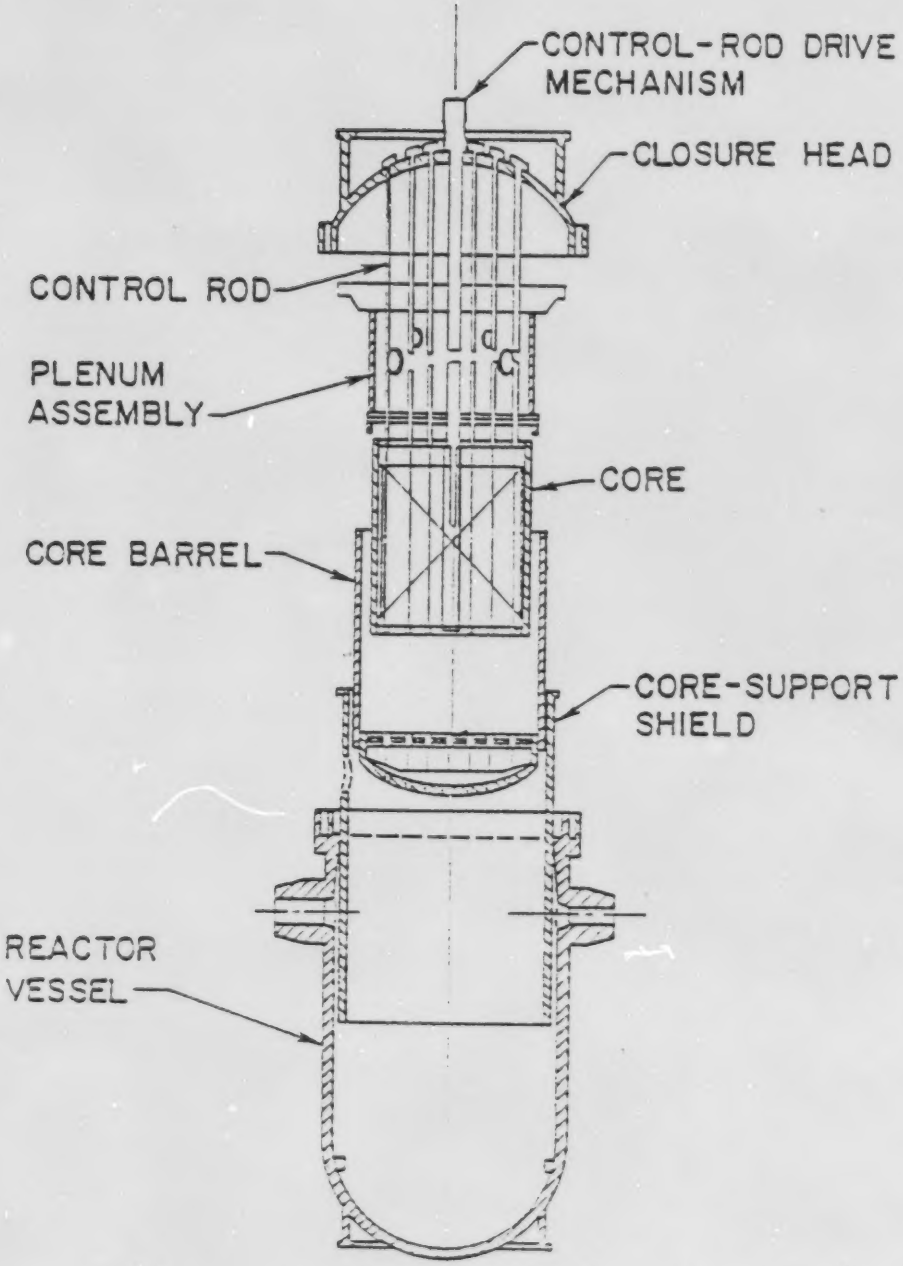

Figure 5-2-2. Typical reactor vessel components.

REACTOR COOLANT PIPING

The reactor coolant (RC) piping interconnects the following components of the reactor system: the reactor, steam generator, pressurizer and reactor coolant pumps (Figure 5-2-3).

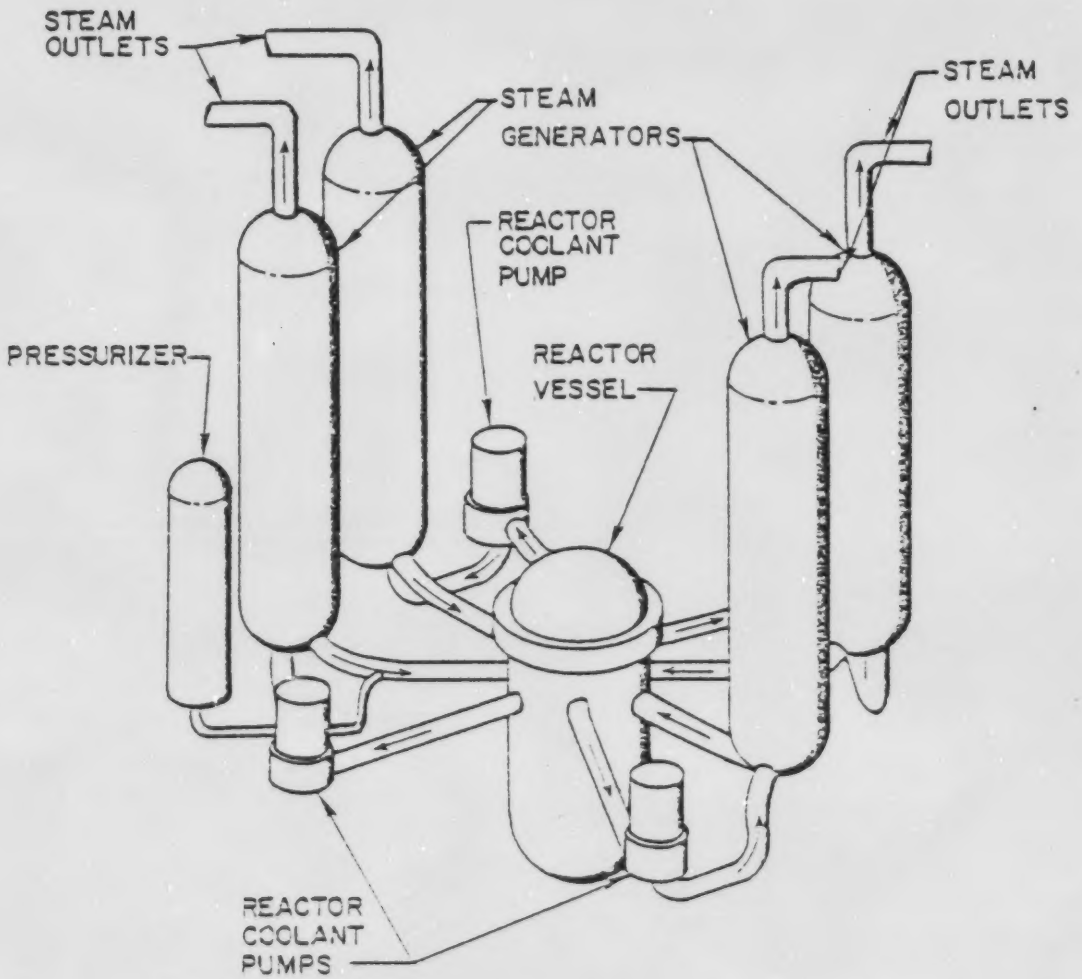

Figure 5-2-3. Typical reactor coolant piping arrangement.

Interconnecting piping, between steam generators and pumps and reactor vessel, is arranged as compactly as possible to prevent excessive heat losses and reduce volume of material subject to possible failure.

The reference point for normal operating pressure is the top of the core. The maximum normal operating pressure is reached at the RC pump discharge. Total pressure at this point consists of pump discharge pressure plus static head pressure from pressurizer water level to elevation at pump, less any losses from pump to top of core.

Piping is arranged as compactly as possible to minimize volume of reactor coolant and reduce energy released during hypothetical accident conditions. An offset in the pump discharge line prevents siphoning of coolant from reactor during refueling or maintenance.

Reactor coolant piping and fittings are either stainless steel, or carbon steel clad with stainless steel. All valves, piping and equipment that operate at elevated temperatures are covered with thermal insulation to reduce heat losses.

All valves in contact with reactor coolant are constructed, primarily, of stainless steel, or employ special materials, such as corrosion-resistant, hard surfacing and packing. Valve designs employ back seats and stem leakoffs to reduce leakage to essentially zero.

PUMPS AND MOTORS

A reactor coolant pump (Figure 5-2-4) usually is a vertical, constant-speed, shaft-sealed unit. The motor, mounted above the pump, is connected to it by a removable shaft coupling,

to facilitate repair without access to coolant, or reduce possible pump contamination if a seal fails.

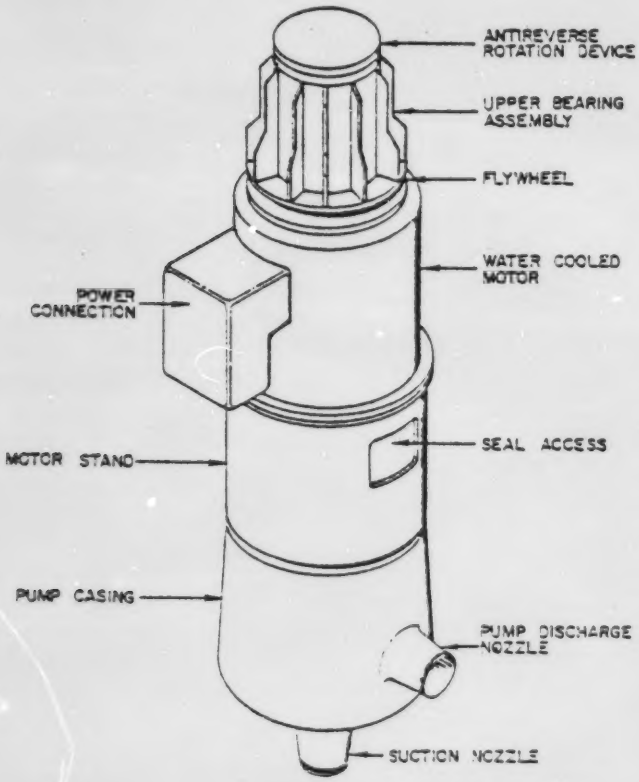

Figure 5-2-4. Reactor coolant pump.

Shaft sealing is accomplished with multiple-face, mechanical seals. These seals are arranged in tandem so that pressure drop between any two seal cavities is always the same. Pressure drops are distributed equally, and multibarrier and redundancy are added into the design. Each seal is designed for full differential pressure between reactor coolant and atmosphere, but reduction of this pressure differential through the use of multiple seals greatly increases reliability and seal life. Pump operation may be continued with loss of

either cooling water or seal injection water, but not with loss of both.

Reactor coolant pump motors are vertical, solid-shaft, squirrel-cage induction devices. Component parts and insulation are designed for at least forty years of continuous service (the average life expectancy of a nuclear power plant). The motor is usually open, drip-proof and air-cooled. An anti-reverse device ensures positive oil films on thrust bearings during startup and shutdown. This device prevents reverse rotation of an idle pump when other pumps in the system are operating.

The thrust bearings have built-in pumping, which eliminates the need for an accessory pump, under normal operating conditions. A flywheel increases system inertia and ensures long pump coastdown time. This provision, together with the reactor control and protection system, ensures adequate flow through the reactor core in the event of a loss of power to the pumps. (Flywheel material is subjected to extensive inspection and examination in accordance with ASME Boiler and Pressure Vessel Code, Section III.)

Accessory devices, included to warn of adverse conditions and to allow equipment monitoring, include:

Motor winding.
Bearing and air temperature detectors.
Oil pressure and level indicators.
Vibration alarms.
Winding differential and surge protection.
Rotation switches.
Water leak detectors.
Space heaters.

These over-design, high safety margins and alarms indicate any impending or actual malfunction out of conservative design

limits.

PRESSURIZER

In an RC system pressurizer (Figure 5-2-5), an electrically-heated pressurizer establishes and maintains reactor coolant pressure within prescribed limits. Provision for a surge chamber and a water reserve to accomodate changes in reactor coolant volume during operation are evident. The unit, a vertical cylindrical vessel, is connected to the reactor outlet piping by surge piping. Two ASME Code relief valves are mounted on top of the pressurizer to relieve any system overpressure. Each valve has half the required relieving capacity. Lifting frequency of Code relief valves is limited by an additional pilot-operated relief valve. The relief valves discharge to a pressurizer quench tank in the reactor building. The quench tank has a stored water supply to facilitate condensation of steam.

A pressurizer has replaceable electric heaters in its lower section and a water spray nozzle in its upper section to maintain steam and water at a saturation temperature corresponding to desired RC system pressure. During reactor pressure decreases, some of the water in the pressurizer flashes to steam to maintain pressure. Electric heaters are actuated to restore normal operation to pressure range. During reactor pressure system increases, steam is condensed by water spray from a reactor inlet line, thus reducing pressure. Spray flow and heaters are automatically controlled by pressurizer pressure controller.

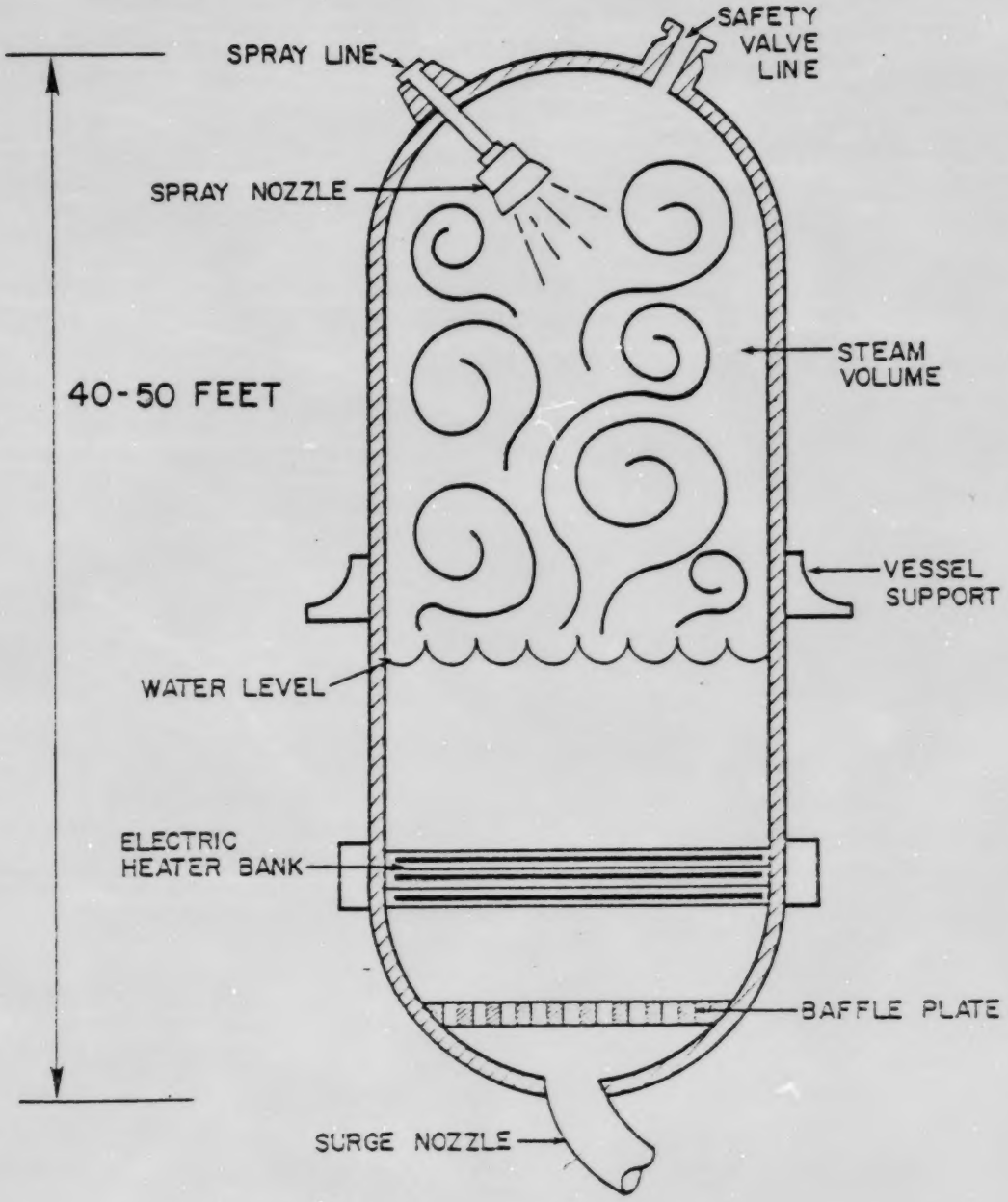

Figure 5-2-5. Typical pressurizer.

STEAM GENERATORS

Two general steam generator designs are manufactured for use with PWR systems:

First: U-tube generators, by Combustion Engineering and by Westinghouse Electric Corporation (Figure 5-2-6). Feedwater entering the feedwater inlet is distributed throughout the periphery of the steam generator by a manifold. This feedwater is heated to saturation temperature, then water and steam mixture enters the moisture separator section where most of the entrained water is removed from the steam. Steam dryers increase steam quality (dryness) more than 99.7% (0.3% moisture).

Second: Once-through steam generator (OTSG), by Babcock and Wilcox (Figure 5-2-7).

This vertical, straight tube-and-shell heat exchanger produces superheated steam at constant pressure over the full power range. Reactor coolant flows downward through the tubes, and steam is generated on the shell side. The shell, the outside of the tubes and the tube sheets form boundaries of the steam-producing section of the vessel. Within the shell, the tube bundle is surrounded by a thin-walled, baffle-plate cylinder, which properly channels secondary-side fluid flow.

Reactor coolant water enters top hemispherical head from reactor and passes vertically downward, through tubes, to lower head and then to RC pumps.

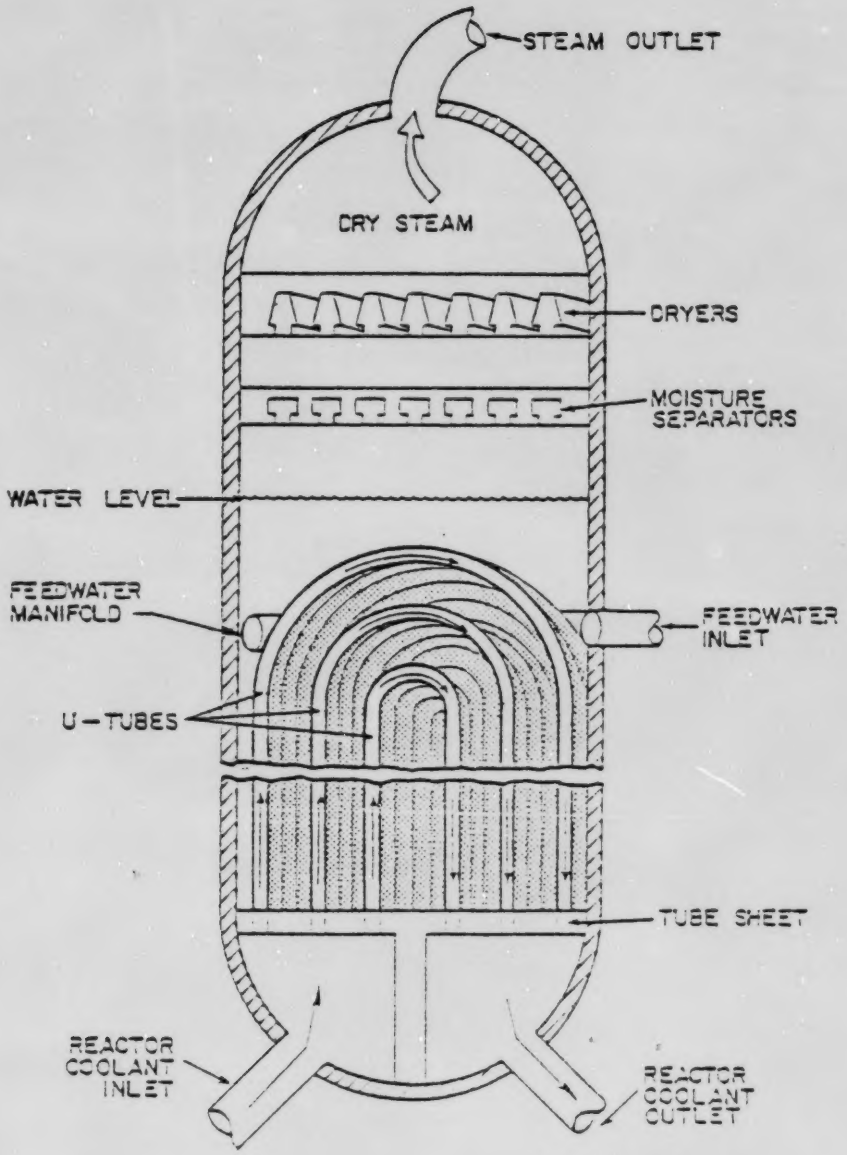

Figure 5-2-6. U-tube steam generator.

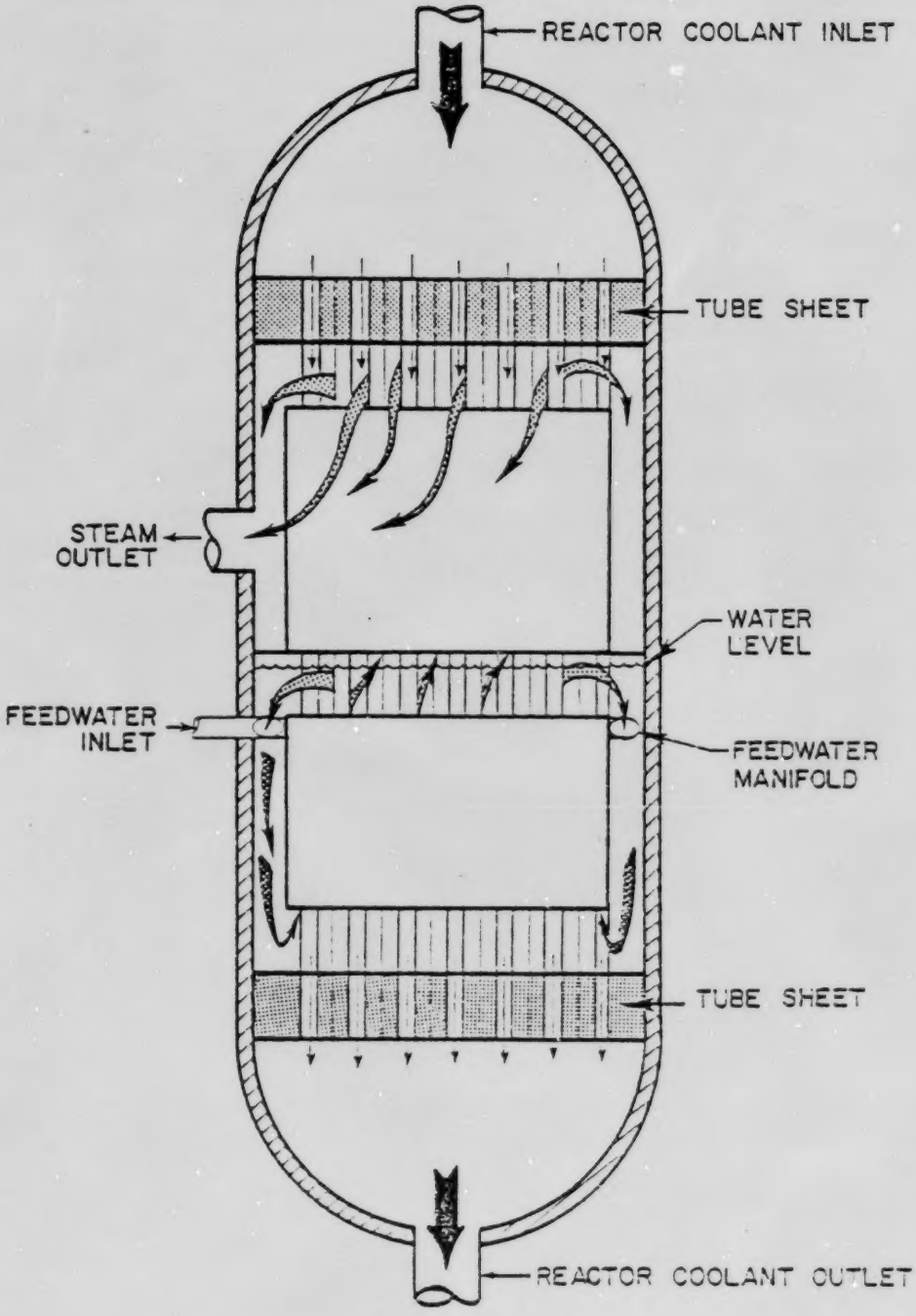

Figure 5-2-7. Once-through steam generator.

CONTROL-ROD DRIVE MECHANISMS

The control-rod drive mechanism (CRDM) varies according to each vendor's design; however, in principle, each is a magnetic device that when de-energized, during a trip, drops the associated control rod into the core to guarantee safe shutdown of the reactor.

One type of CRDM is the magnetic jack (Figure 5-2-8). It consists of a pressure housing, a magnetic coil, latch assembly, position indicator and control-rod drive shaft. Magnetic coils, installed outside the pressure housing, have provisions for air cooling.

The gripper latch and armature are inside of the pressure housing. The gripper latch engages a notched (serrated) rod This notched rod can be jacked up or down in small increments by sequential stepping of the set of magnetic coils. A trip action occurs when the magnetic coils are de-energized, allowing the rod to drop into the core.

During normal operation, only one magnetic coil is energized to hold the latch on the drive shaft. Position indication is by a series of differential transducers mounted to indicate the position of the drive shaft relative to the magnetic coil.

Other types of control-rod drive mechanisms are described in detail in Module 5-1, "Basic Nuclear Reactor Components."

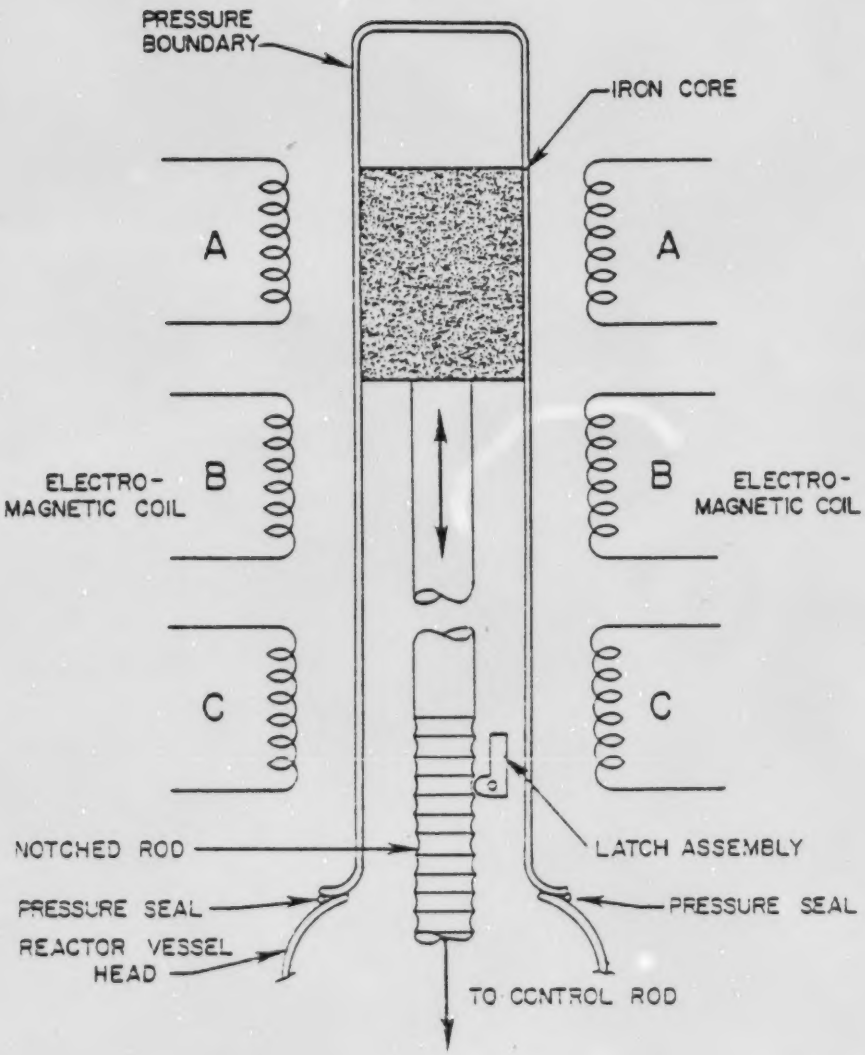

Figure 5-2-8. Magnetic jack control-rod drive.

CONTROL-ROD ASSEMBLIES

Reactor startups, load change requirements and control of changes in reactivity are affected by movement of control-rod assemblies (Figure 5-2-9). These assemblies consist

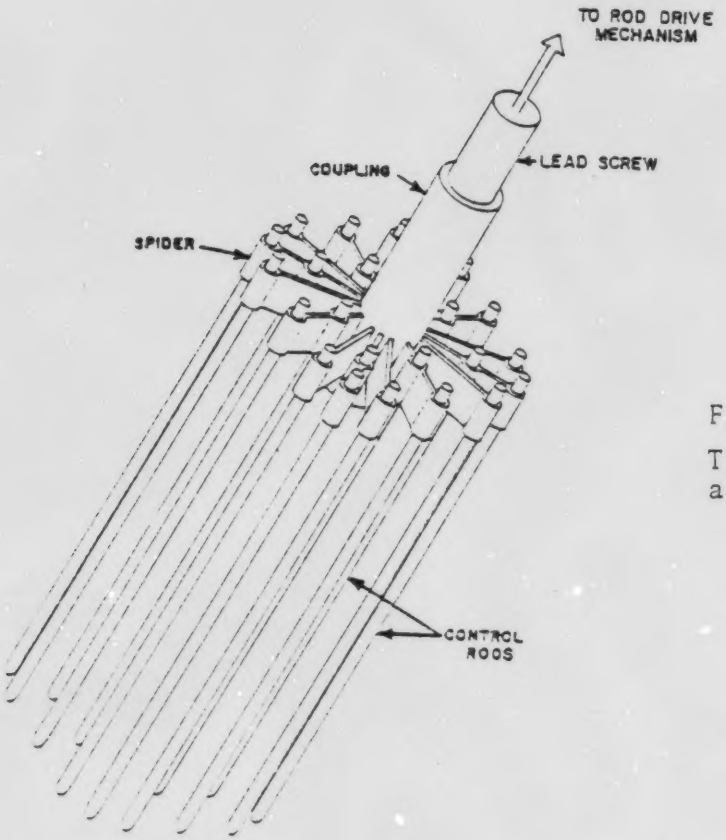

Figure 5-2-9.
Typical control-rod assembly.

of cylindrical rods positioned inside the fuel assembly in vacant positions unoccupied by fuel rods. The control rods are inserted through the upper end of the fuel assembly. Full-length guidance is provided by guide tubes. Each assembly is coupled to the control-rod drive assembly by a separate

mechanism mounted on the reactor vessel head.

Number and design of control rods varies from vendor to vendor. Each uses different neutron absorbers; e.g., some use boron carbide; others use a combination of silver, indium and cadmium as poison (absorber) material.

Axial power shaping rod assemblies, used to control axial power distribution in the core, are similar to control-rod assemblies, except that the neutron absorbing material is located only in the bottom three feet of the rods.

Burnable poison rod assemblies, which are stationary, achieve improved neutron economy. Most PWR designs have burnable poison in the initial core for the first cycle of operation only.

CHEMICAL ADDITION AND BORON RECOVERY SYSTEMS

Another method of controlling reactor power is the addition of boric acid to reactor coolant. Boron readily absorbs thermal neutrons as it passes through the reactor core. By controlling the concentration of boric acid in the coolant, reactor power is maintained, or altered. This process, known as "chemical shim," is used during normal operation.

Boric acid solution also can be injected during non-operating periods to increase the margin of shutdown, thereby reducing the probability of an inadvertent criticality.

SECONDARY LOOP

The secondary loop (Figure 5-2-10) transports steam

(thermal energy) from steam generators to turbines.

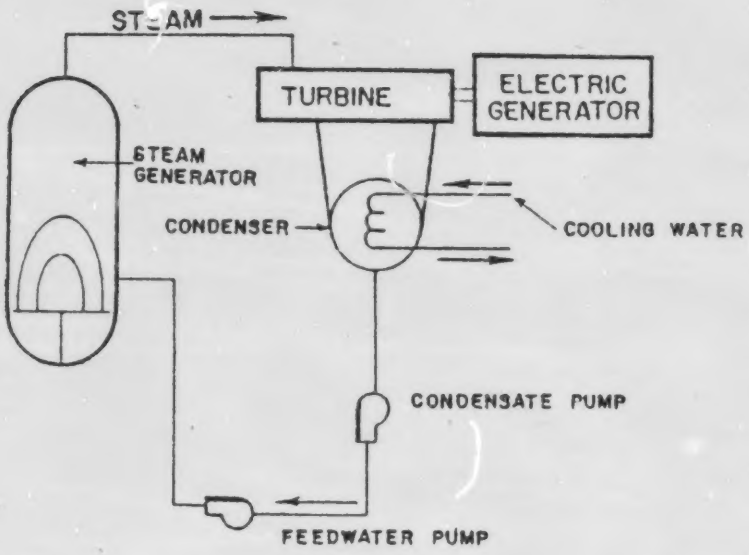

Figure 5-2-10. Secondary loop.

When steam leaves the steam generator, it flows through steam piping to the turbines. Portions of the steam may be diverted to supply power to auxiliary equipment. As the steam drives the turbines, thermal energy is reduced considerably and, as it exits the turbine, a water-cooled condenser further reduces the energy, causing the steam to condense and collect in a storage area (condenser "hotwell").

From this hotwell, water (condensate) passes to the feed pumps, which inject the feedwater back into the steam generator to be converted into steam, and the cycle reoccurs.

EXERCISES

1. State function of the following:
   - Reactor coolant pumps.
   - Pressurizer.
   - Chemical addition and boron recovery system.
2. Sketch (schematic diagram) a typical PWR system; include secondary loop.
3. Briefly describe operational differences between a U-tube type and a once-through type steam generator; make a simple sketch of each.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

American Society of Mechanical Engineers (ASME). ASME Boiler and Pressure Vessel Code, Section III. New York: ASME, 1977.

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Tipton, C. R. Reactor Handbook I - Materials. New York: Interscience Publishers, Inc.; 1960.

1. Sketch a PWR vessel; show major components and coolant flowpath:

2. Briefly explain design and operating characteristics of a typical reactor coolant pump:

3. Sketch a typical PWR pressurizer: describe its operation.

4. Sketch and explain operation of both types of PWR steam generators:

5. Sketch and explain operation of two types of PWR control-rod drive mechanisms:

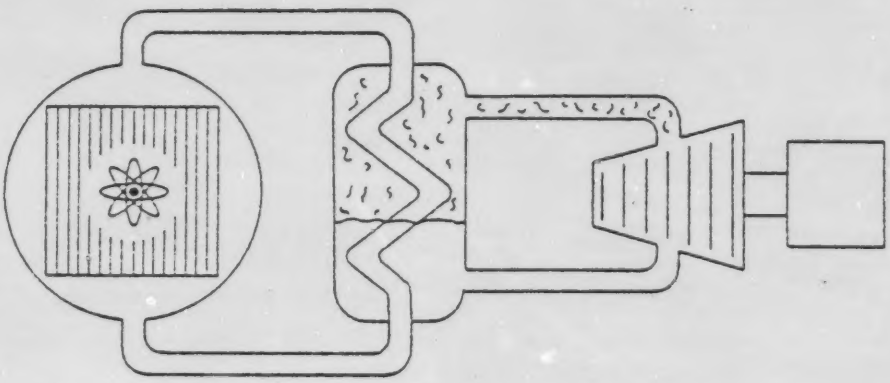
6. Why is boron added to reactor coolant?

7. Sketch and briefly explain operation of a typical secondary loop:

NUCLEAR TECHNOLOGY

MODULE 5-3 BOILING WATER REACTORS (BWR). -

GEORGE J. STOLL, P.E.



INTRODUCTION TO NUCLEAR SYSTEMS

COURSE 5

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST

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COURSE 5 INTRODUCTION TO NUCLEAR SYSTEMS

MODULE 5-1 Basic Nuclear Reactor Components

MODULE 5-2 Pressurized Water Reactors (PWR)

MODULE 5-3 Boiling Water Reactors (BWR)

MODULE 5-4 Primary Cooling Systems

MODULE 5-5 Nuclear Instrumentation

MODULE 5-6 Nuclear Fuel Cycle

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INTRODUCTION

The boiling water reactor (BWR) concept of nuclear steam generation plants claims the following advantages:

No intermediate heat exchangers.

Turbine throttle temperature about the same as reactor outlet temperature.

Simple design.

Shorter construction schedules.

Lower primary flow (no secondary loop).

Disadvantages of the system include 1) its inherent carryover of radioactivity (via steam) to the turbine and other equipment, 2) its attendant maintenance precautions and 3) its difficulty in maintaining constant power because of the effects of void volume and flow changes on reactor power.

MODULE PREREQUISITES

The student should have completed Module 5-2, "Pressurized Water Reactor (PWR)," and its prerequisites.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Describe typical operating conditions, including temperatures, pressures and powers.
2. Given an appropriately labeled drawing, describe purpose of each principal component of a BWR nuclear steam system.
3. Given an appropriate diagram, discuss and label flow paths inside vessel and describe purpose of jet pump.
4. State where control rods enter reactor vessel, why they enter at this location and how they are fail-safe.
5. Define:
   - Moisture separator.
   - Feedwater heater.
   - Jet pump.
   - Steam control valve.
   - Cruciform.
6. Sketch and explain operation of a BWR moisture separator system.
7. Sketch and explain a typical BWR fuel assembly and control rod.
8. Explain basic operation of a typical BWR control-rod drive system.

BOILING WATER REACTORS

The popularity and simplicity of the boiling water reactor (BWR) system is evident: about half the light-water reactors used in the United States are of this type.

Water boils in the reactor from heat produced in the core (Figure 5-3-1); wet steam (i.e., contains water drop-

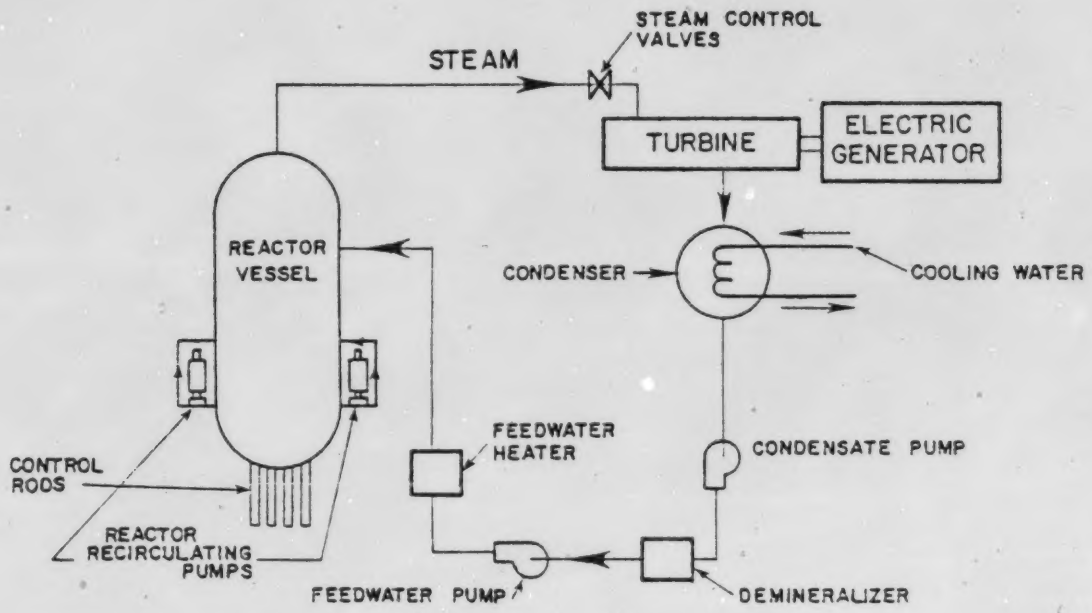

Figure 5-3-1. BWR system.

lets) is dried in the moisture separators and dryers, situated in the top of the reactor vessel. Steam then goes to the turbine, where its energy is converted to mechani-

cal energy to drive an electric generator. Used steam, changed back into water in the condenser, is pumped through demineralizers and feedwater heaters, back to the reactor.

Part of the total water flow through the system is recirculated for reactor control purposes (Figure 5-3-2). The

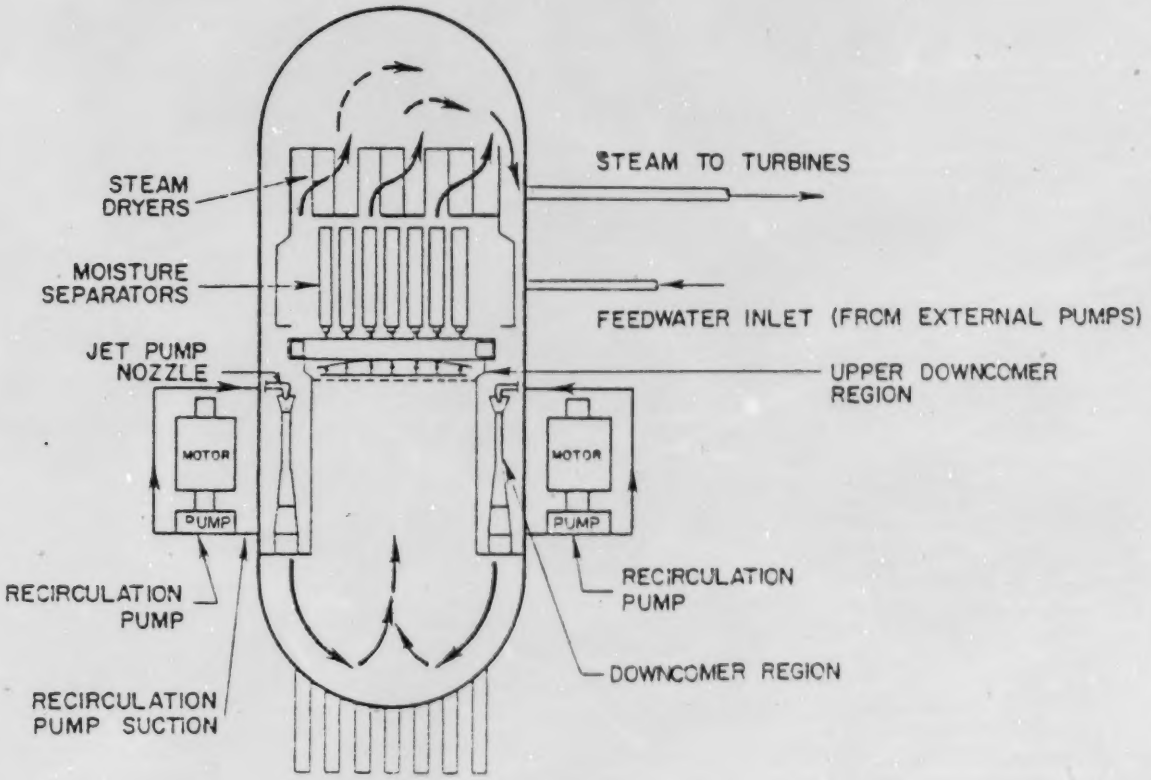

Figure 5-3-2. Steam and recirculation water flow paths.

coolant is recirculated through the reactor core by two external loops and an internal jet pump system. External pumps pull a suction on the downcomer region and discharge

into the jet pump inlet, causing a low-pressure area, which draws water from the upper downcomer region and discharges it into the core region. From this point, water passes through the core region to become steam or to be recirculated.

Natural circulation can sustain the reactor up to approximately one-third power without jet pumps in operation - an important advantage, enabling the plant to start up without external electrical power to operate pumps.

Typical operating conditions for a BWR are given in Table 5-3-1.

TABLE 5-3-1
TYPICAL BWR OPERATING CONDITIONS

Operating reactor pressure	1040 psia
Operating reactor temperature	548°F
Steam flow	15.4×10^6 lbs/hr
Rated power level	3579 MWt
Net electrical output	1220 MWe
Reactor core water flow	105×10^6 lbs/hr

REACTOR ASSEMBLY

A BWR nuclear steam supply system (NSSS) includes:
Reactor vessel and internals.

Recirculation system.

Control-rod drive system.

Steam line.

Fuel and instrumentation.

The reactor vessel (Figure 5-3-3) is a cylindrical

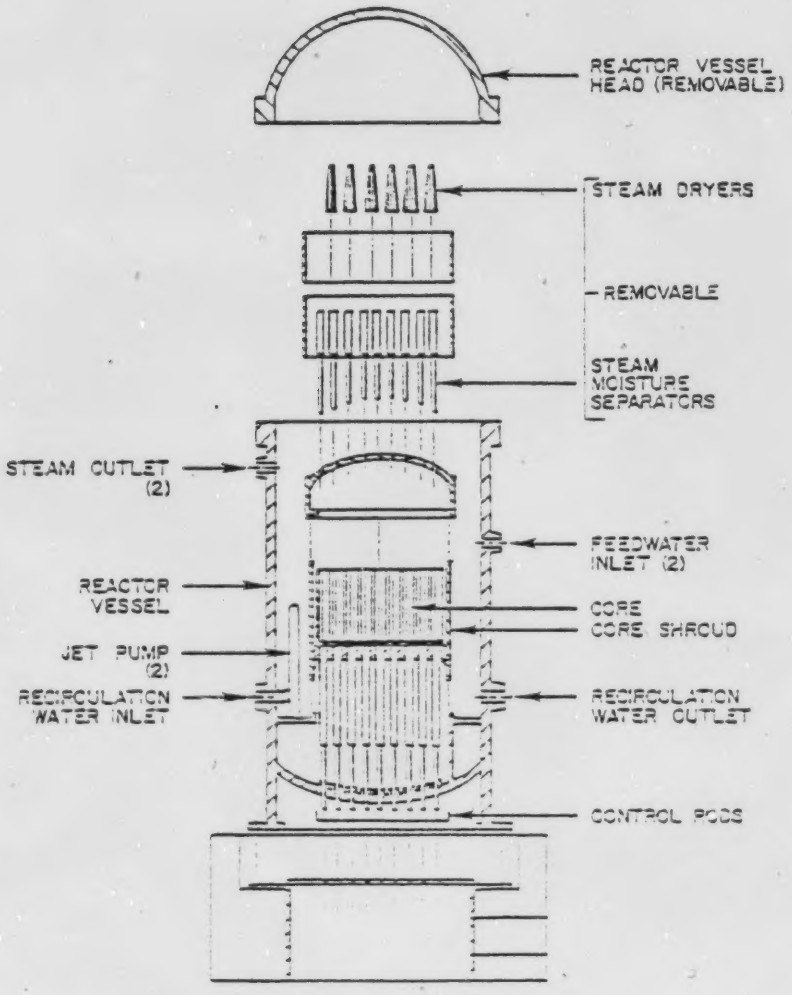

Figure 5-3-3. Reactor vessel and internals.

pressure vessel fabricated of alloy carbon steel, internally clad with stainless steel (to combat corrosion). The removable head is fabricated without cladding because the steam environment is not susceptible to corrosion.

The vessel is mounted on a support skirt, bolted to a concrete and steel cylindrical pedestal built into the reactor building foundation. The removable vessel head is sealed by two concentric metal O-ring gaskets to prevent leakage under all operating and test conditions. A gasket leak detection system monitors seal integrity. The upper portion of the vessel contains steam drying and moisture separating equipment, while the lower portion contains the core (Figure 5-3-3).

The steam moisture separator assembly (Figure 5-3-4)

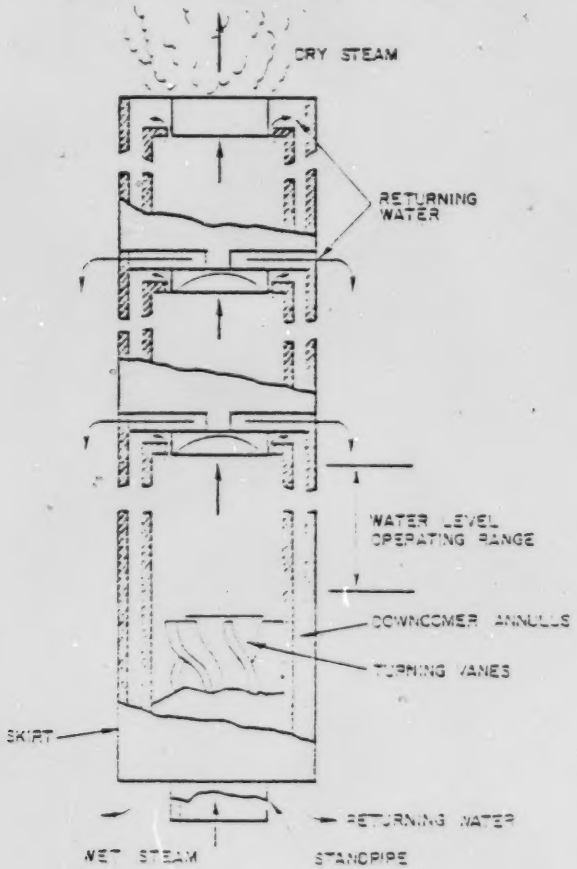

Figure 5-3-4.
Internal steam moisture separator.

consists of a group of pipes containing a 5-stage steam separator.

The steam dryer assembly, in the top of the reactor vessel, consists of a series of vanes (Figure 5-3-5). Moisture

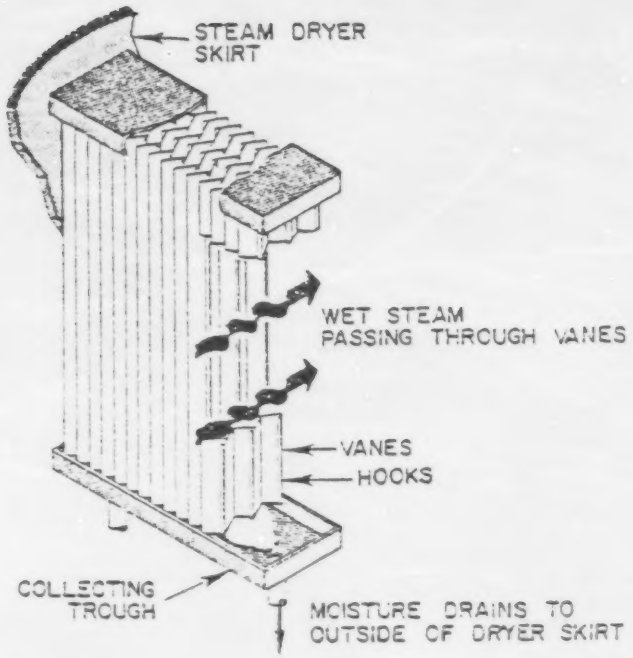

Figure 5-3-5.
Steam dryer.

is removed by changes in direction, collects in troughs, and spills into the downcomer annulus.

Steam flowing to the turbines is controlled by a steam control valve in the main steam lines. As reactor power is increased by any method, the turbine senses an increase in pressure and signals the steam control valve to open,

permitting more steam to pass into the turbine, increasing output.

REACTOR CORE

BWR core structure, configuration and fuel element design have remained substantially the same throughout the plant design evolution (from the 500-MWe period to the present 1100- to 1200-MWe plants).

The reactor core (Figure 5-3-6), a circular lattice

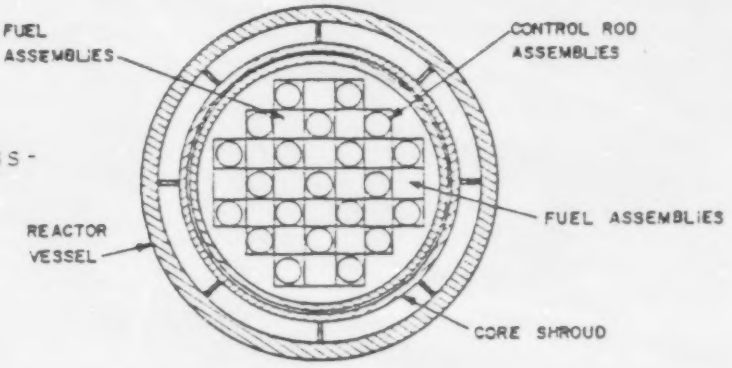

Figure 5-3-6.
BWR core plan (cross-sectional view).

arrangement containing fuel assemblies, is situated in the reactor vessel. Core design is such that it has sufficient margin to accept changes and transient operation without damage.

FUEL ASSEMBLIES

A BWR fuel assembly (Figure 5-3-7) has 64 fuel rods

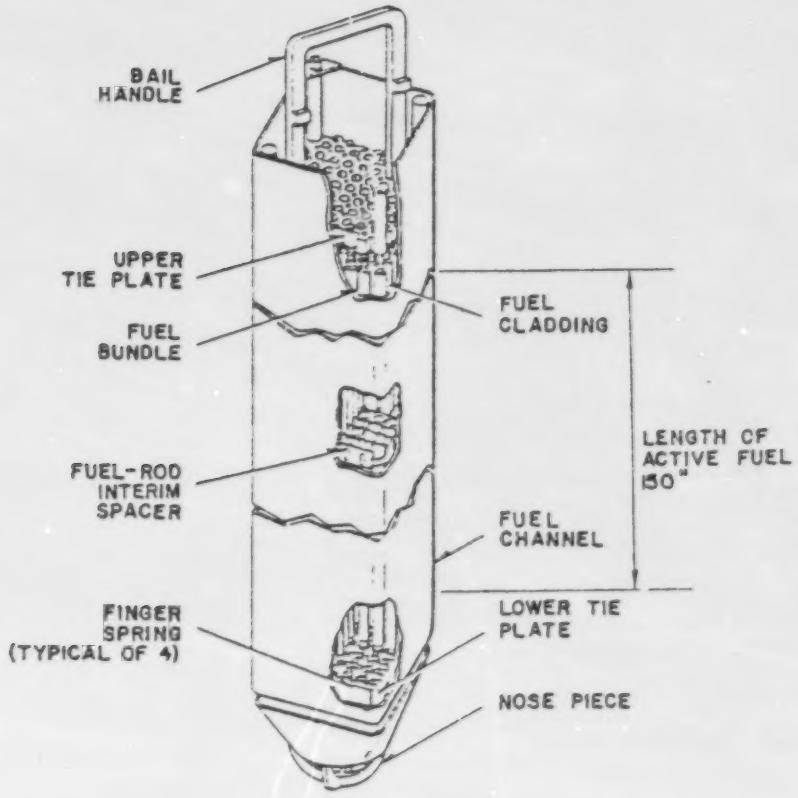

Figure 5-3-7. BWR fuel assembly.

(8 by 8). It is supported by lower and upper tie plates (made of stainless steel) and is surrounded by a metal channel made of high-strength, corrosion-resistant material (Zircaloy) with low neutron absorption.

Each fuel rod (Figure 5-3-8) is composed of a Zircaloy tube in which fuel pellets are stacked. The rod is gas-filled and sealed (by welded plugs in each end). At the top end of the rod, a cavity contains a neutron poison gas plenum and spring to keep pellets in place during irradiation of the fuel assembly.

The channel surrounding each fuel assembly eliminates cross-flow between assemblies, simplifies calculations nec-

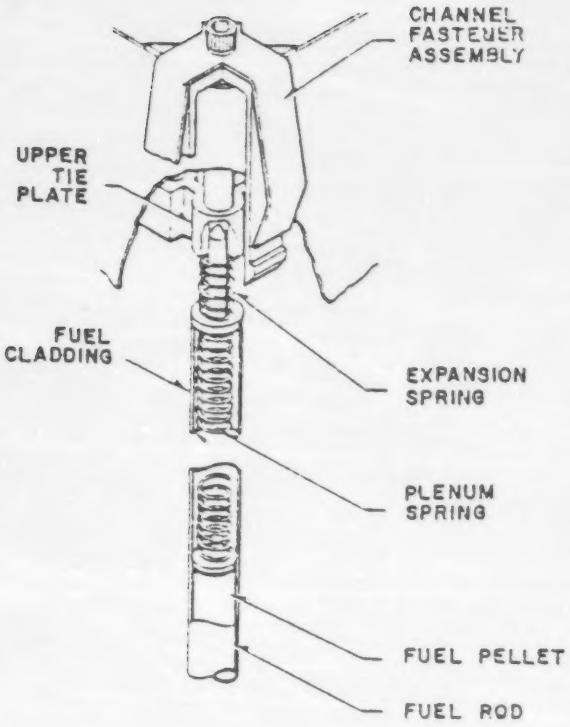

Figure 5-3-8.
BWR fuel rod.

essary for thermal-hydraulic design, and directs positioning of control rods.

CONTROL RODS

Control rods (Figure 5-3-9) contain stainless steel tubes filled with compacted boron carbide powder in stainless steel envelopes (sheaths). Each sheath contains 84 tubes, 21 in each wing. The cross-sectional shape of the control rod is called a cruciform. Positioning rollers at the top and bottom of the sheath aid in providing guidance for insertion and withdrawal of the control rod. The bottom of the control rod is in the configuration of a bell

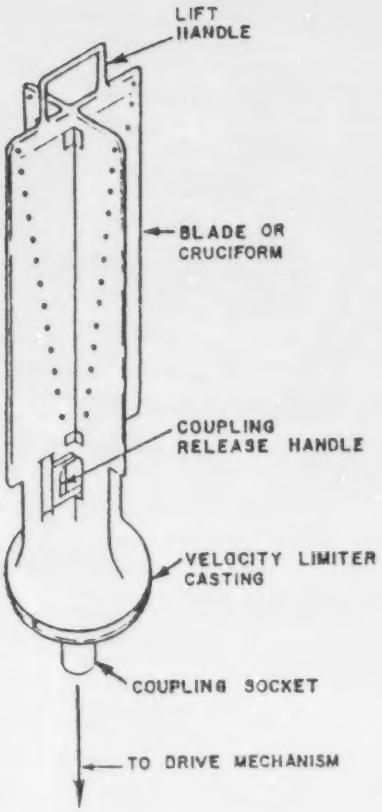

Figure 5-3-9.
BWR control rod.

or parachute to limit velocity of drop. Holes permit water to flow through the assembly and cool the rod.

Adequate shutdown and fail-safe capability at all times is designed in the control rod system, with the capability of keeping the core subcritical with any single control rod in a fully withdrawn position. Withdrawal rates are limited to prevent fuel damage.

CONTROL-ROD DRIVE SYSTEM

Control-rod drives, in a BWR system, enter the reactor vessel upward through the bottom head in a powered shutdown

made against the force of gravity. Control rods enter through the bottom of the vessel to avoid the steam separators and dryer located in the top of the vessel. Also, control-rod drive mechanisms are accessible for inspection and service, and they do not interfere with refueling. Yet another reason for bottom entry is that the rods are in the liquid region where they are most effective, not in the upper steam region, where they have little reactivity worth.

A BWR control-rod drive system is a complicated set of hydraulic lines and valves in which rods are moved by pressure differences (between the pressure vessel and the hydraulic system). Detailed discussion of this system is deferred to Course 9, "Reactor Auxiliary Systems."

JET PUMPS

Each jet pump consists of five basic parts:

Inlet nozzle.
Driving nozzle.
Suction inlet.
Mixing assembly.
Diffuser.

Jet pump operation (Figure 5-3-10) can be described as follows: driving flow is constricted by a nozzle; velocity increases and pressure decreases, creating a low-pressure region at inlet suction of jet pumps, drawing surrounding water into pump. This water, called "driven flow," makes up about two thirds of the total jet pump and core flow. The driving and driven flows are mixed (by a mixing assembly) to reduce possible temperature variations. The flow then passes to the diffuser. There, fluid velocity

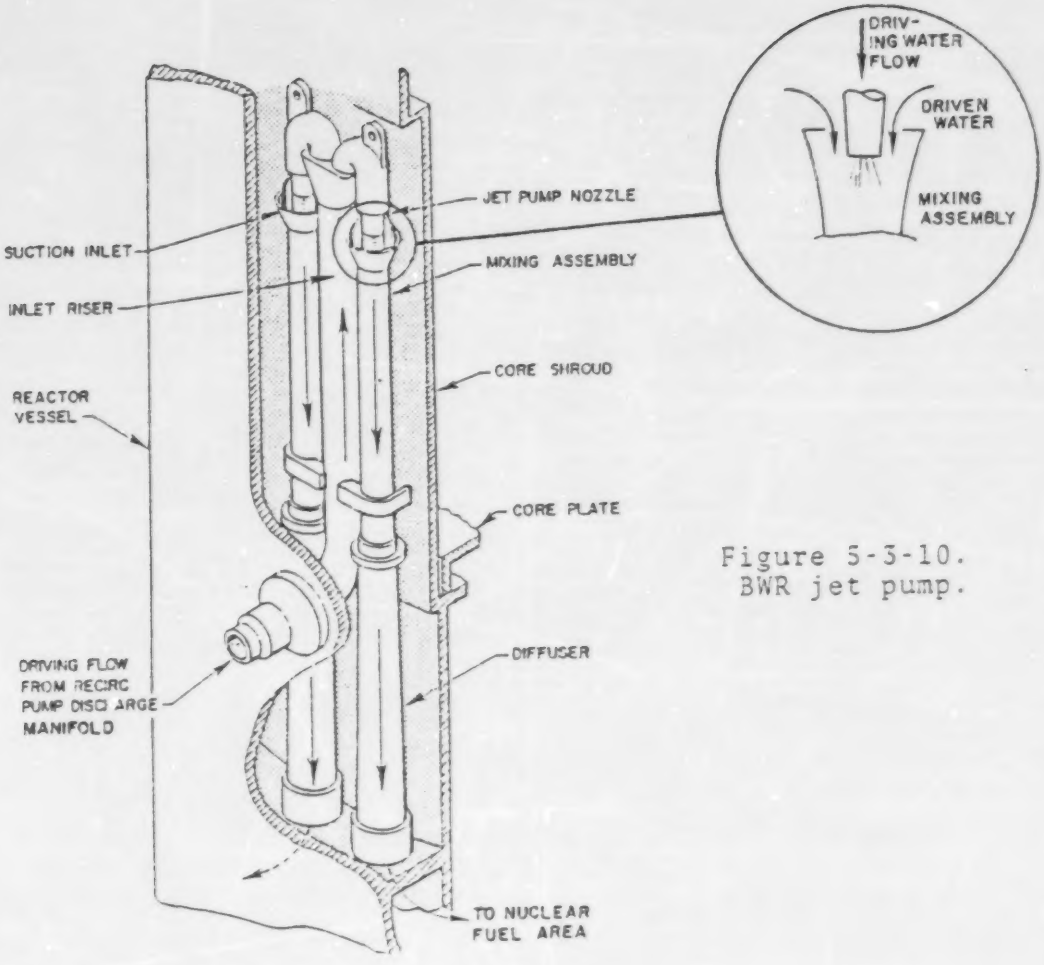

Figure 5-3-10.
BWR jet pump.

decreases, resulting in a pressure increase, which forces water into lower core region and up into core.

In a typical BWR recirculation jet pump (Figure 5-3-11), driving flow (from recirculation pump discharge) flows up inlet riser and down through jet pump nozzles. From the diffuser, it passes up through the core.

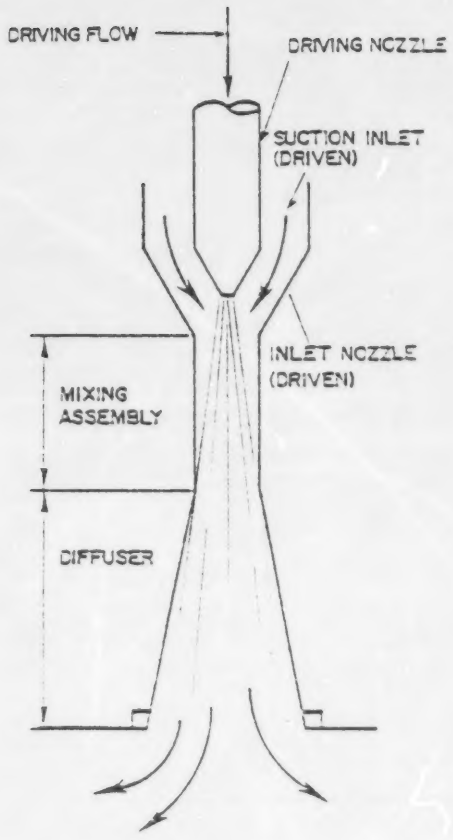

Figure 5-3-11.
Jet-pump principle.

FEEDWATER HEATERS

The purpose of a feedwater heater is to preheat condensate from the condenser before it is returned to the reactor vessel (Figure 5-3-12). As preheat is provided by extract steam from the low-pressure turbines, it helps overall efficiency of plant by reducing the amount of heat required by the reactor to boil water.

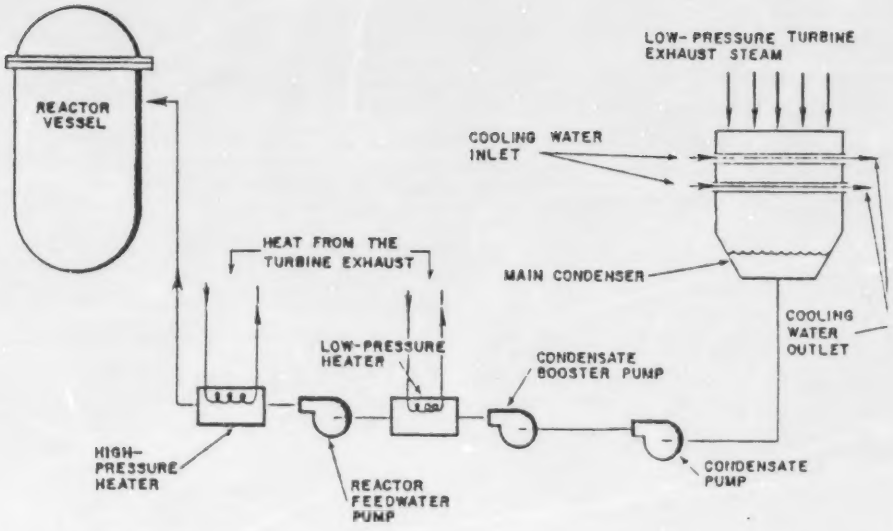

Figure 5-3-12. Condensate and feedwater system.

EXERCISES

1. List advantages claimed for a BWR system.
2. Sketch and label principal components of a BWR system.
3. State where BWR control rods enter reactor vessel and how they are fail-safe.
4. Describe fluid (including steam) flow through a BWR system.
5. Describe method of separating moisture from steam.

MATERIALS

None.

PROCEDURES

1. Obtain a Nuclear Buyer's Guide; select manufacturers of BWR:
   - Jet pumps.
   - Moisture separators and steam dryers.
   - Control-rod drive systems.
2. Instructor will provide descriptive manuals for BWR power plants, individual systems and/or components, such as jet pumps, moisture separators and steam dryers and control-rod drives. Make oral report to class, describing assigned system or components.

DATA TABLE

None.

REFERENCE MATERIALS

American Nuclear Society. Nuclear Buyer's Guide. Hinsdale, IL: American Nuclear Society, annual publication.

General Electric (GE). General Description of a Boiling Water Reactor. San Jose, CA: GE, Nuclear Energy Division, April 1975.

Lish, Kenneth C. Nuclear Power Plant Systems and Equipment. New York: Industrial Press, 1972.

Tipton, C.R. Reactor Handbook. New York: Interscience Publishers, Inc., 1960.

Van Ness, H.C. Understanding Thermodynamics. New York: McGraw-Hill Book Co., 1969.

1. Sketch and briefly explain a typical BWR system:

2. Sketch and briefly explain a BWR vessel; describe components and flow paths; include detailed description of recirculation:

3. List five components of a BWR nuclear steam supply system:

(1)

(2)

(3)

(4)

(5)

4. Sketch and briefly explain operation of a BWR moisture separator and steam dryer:

5. Sketch and briefly explain a typical BWR fuel assembly:

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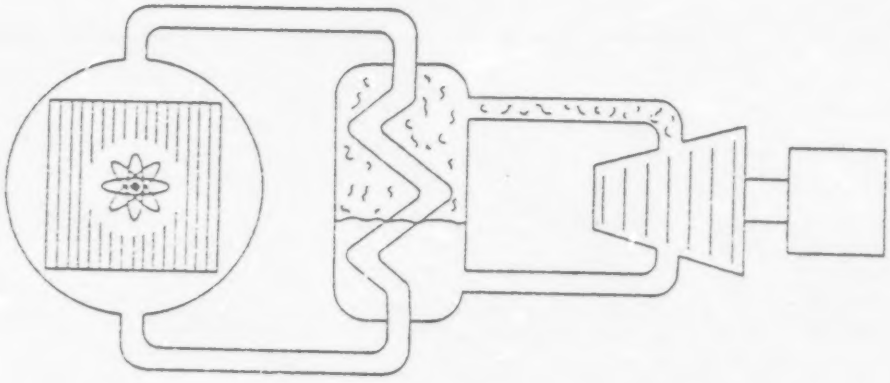
6. Sketch and explain a typical BWR control rod:

7. Briefly explain a BWR control-rod drive system:

NUCLEAR TECHNOLOGY

MODULE 5-4 PRIMARY COOLING SYSTEMS. -

GEORGE J. STOLL, P.E.



INTRODUCTION TO NUCLEAR SYSTEMS

COURSE 5

TEC-SW 83

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST

4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 5 INTRODUCTION TO NUCLEAR SYSTEMS

MODULE 5-1 Basic Nuclear Reactor Components

MODULE 5-2 Pressurized Water Reactors (PWR)

MODULE 5-3 Boiling Water Reactors (BWR)

MODULE 5-4 Primary Cooling Systems

MODULE 5-5 Nuclear Instrumentation

MODULE 5-6 Nuclear Fuel Cycle

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INTRODUCTION

A nuclear steam system consists of a reactor vessel, steam generators, heat exchangers, piping, valves and other equipment needed to extract heat from the core. These components, when linked together, become the reactor coolant system. Many of these components are discussed in the last two modules. This module describes those items common to both PWR's and BWR's, including piping, valves, insulation, design specifications and inspections.

MODULE PREREQUISITES

The student should have completed all previous modules in this Course, and their prerequisites.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Identify appropriate specifications for marking primary system valves.
2. Explain how nuclear valves differ from valves for conventional purposes.

3. List materials used for BWR primary system liquid piping; explain how it differs from PWR primary system piping.
4. List materials and basic requirements for primary pipe insulation.
5. Define isolation valve.

PIPING

In a BWR system, carbon steel is used for the primary steam piping. In a PWR system, the piping is stainless steel or carbon steel clad with stainless. Most piping operating at high temperatures is insulated to reduce heat losses.

Reactor coolant piping provides the interconnecting link between individual components of a reactor coolant system. The piping and components are arranged as compactly as practical to reduce total energy stored in the system and, thus, the energy released in the event of a major coolant system rupture.

Generally, reactor coolant piping is constructed of one of the following materials:

Carbon steel.

Carbon steel clad (covered) internally with stainless steel.

Stainless steel.

Regardless of the material used, it must meet rigid standards pertaining to quality, construction and testing, as set forth by the ASME Boiler and Pressure Vessel Code and Appendix B of 10CFR50.

PRIMARY SYSTEM VALVES

All nuclear primary system valves are built to the ASME

Code, Section III, Code Class I, Seismic Category I, Quality Group A, and are of cast or forged stainless steel, depending on size.

BWR system safety-relief valves are dual function, discharging directly to the pressure suppression pool. Their safety function includes protection against overpressure of the reactor primary system, in accordance with ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Vessels," Article 9.

These valves open at "spring set point pressure" and close when inlet pressure falls to 96% of that spring set point pressure. They are power actuated from the control room or, automatically, by too high pressure.

Isolation valves for the main steam line are spring loaded, pneumatic-piston-operated, globe valves designed to close on loss of pneumatic pressure or loss of power to the pilot valves. Once isolation is initiated, valves continue to close and can be opened only by manual means. These valves are carbon steel, ASME SA-352, Type LCB, or SA-216, Type WCB.

Shutoff and bypass valves in the recirculation loop are motor-operated gate valves. Each flow control valve is a ball type, actuated electro-hydraulically (on the discharge side of a pump). Fabrication requirements for these valves are the same as for the attached piping.

Valves for PWR systems in contact with reactor coolant are of stainless steel, or contain hard surfacing and packing.

PWR safety valves are self-actuating and spring-loaded with power-operated, automatic relief valve backups. If excessive leakage occurs, these relief valves are isolated by remotely-controlled stop valves.

Gaskets are of either spiral wound stainless steel or solid metal. Valve stems are packed with material essentially impervious to radiation damage.

Valve operators, devices that remotely move valve stems, in nuclear systems, are usually one of two types: electric or pneumatic.

INSULATION

All valves, piping and other equipment operating at high temperatures are insulated to reduce heat losses. (Asbestos use has been severely curtailed by Occupational Safety and Health Act (OSHA) requirements.) Other insulation, though more susceptible to damage when removed for inspection or maintenance, is used.

Reflecting type insulation (Figure 5-4-1) is widely

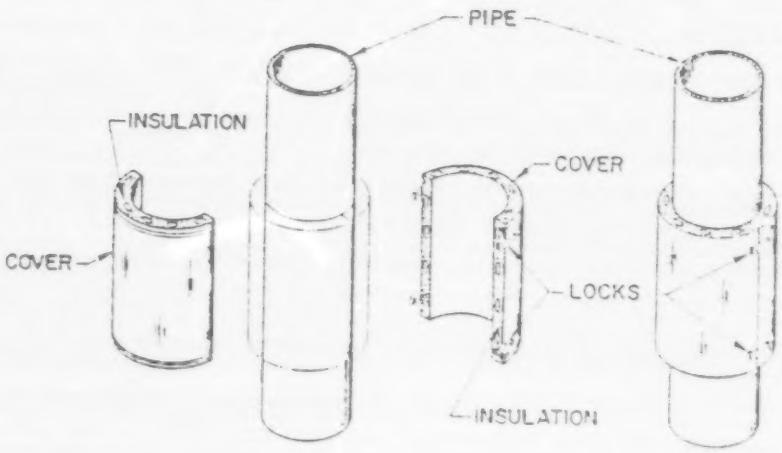

Figure 5-4-1. Reflecting type insulation.

used in modern plants; it consists of stainless steel, or a combination of stainless steel and aluminum. Primarily, stainless steel is used with PWR's and stainless and aluminum composite is used with BWR's.

Advantages of these insulation materials include:

Excellent thermal properties.

Contains no fibers to absorb atmospheric moisture, condensation or other fluids that could cause corrosion.

Removal and reinstallation times are minimal when in-service inspection is required.

DESIGN SPECIFICATION

Design specifications, which must cover all primary system components, including hardware, maintenance and operating instructions, quality-assurance reports and drawings, are prescribed in a rigid manner.

Law requires a written quality-assurance program from beginning to end, for all phases of nuclear equipment design, fabrication and operation. These criteria are spelled out in Title 10, Code of Federal Regulations, Part 50, Appendix B, entitled, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants."

IN-SERVICE INSPECTION

Public health and safety demand that integrity and reliability of a nuclear power plant be assured by periodic inspections. Section XI of the ASME Boiler and Pressure Vessel Code sets forth the examination requirements for nuclear power plants (an inspection must be performed during annual refueling shutdowns over a 10-year period).

The system, therefore, must be designed to allow access to all equipment for inspection. (Remote inspection equipment must be available for special situations.)

A base-line or prestartup inspection must be made to use as a basis for all subsequent inspections. This base-line inspection involves the primary coolant system.

An annual inspection generally consists of visual, surface and volumetric (internal) examinations. Liquid penetrant and magnetic particle methods may be used for surface examinations, and volumetric examinations may be performed by using radiographic and ultrasonic techniques.

Remote inspection equipment must be used to perform post-operational examinations required by Section XI of the ASME Code in radioactive environments.

In-service inspection techniques are used for volumetric examinations, and closed-circuit television cameras have been designed (for high-radiation environments) to provide visual inspections.

An automated reactor inspection system provides for inspection of circumferential and longitudinal vessel shell welds, as well as inlet, outlet and core-flooding nozzle-to-vessel welds, inlet and outlet nozzle to pipe welds, flange ligament area welds, bottom head-to-vessel welds and bottom

head welds. This system provides computer-controlled data reduction and analysis with a display of ultrasonic indications for review and action.

EXERCISES

None.

MATERIALS

1 copy of ASME Section III Boiler and Pressure Vessel Code.

PROCEDURES

1. Use ASME Section III Boiler and Pressure Vessel Code, Subsection NB; locate valve material specifications and complete Data Table 5-4-1.
2. Visit a radiography facility and watch radiography of a valve body or similar component; write a one-page report describing test.

DATA TABLE 5-4-1
VALVE MATERIAL SPECIFICATIONS

MECHANICAL PROPERTIES CLASSIFICATIONS	APPLICABLE SPECIFICATION NUMBER	
	FORGING	CASTING
Carbon Steel Carbon Steel (low temp) Carbon Moly Carbon Moly (low temp) 1/2 Cr - 1/2 Mo 1 Cr - 1/2 Mo 1 1/4 Cr - 1/2 Mo 2 1/4 Cr - 1 Mo		
Type 304 Type 304L Type 310 Type 347 Type 321 Type 316 Type 316L		

REFERENCE MATERIALS

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American Society of Mechanical Engineers (ASME). ASME Boiler and Pressure Vessel Code. New York: ASME, current.

Babcock and Wilcox Co. Babcock - 205 NSS Design Summary. Lynchburg, VA: Babcock and Wilcox Co., 1976.

General Electric Co. (GE). General Description of a Boiling Water Reactor. San Jose, CA: GE Nuclear Energy Division, April 1975.

Lish, Kenneth C. Nuclear Power Plant Systems and Equipment. New York: Industrial Press, 1972.

Masche, George C. Systems Summary of a Westinghouse Pressurized Water Reactor Nuclear Power Plant. Pittsburgh, PA: Westinghouse Electric Co., 1971.

1. Given the ASME Boiler and Pressure Vessel Code Section III, list specifications for a 4-inch primary system valve:

2. Briefly explain how nuclear valves differ from valves for conventional purposes:

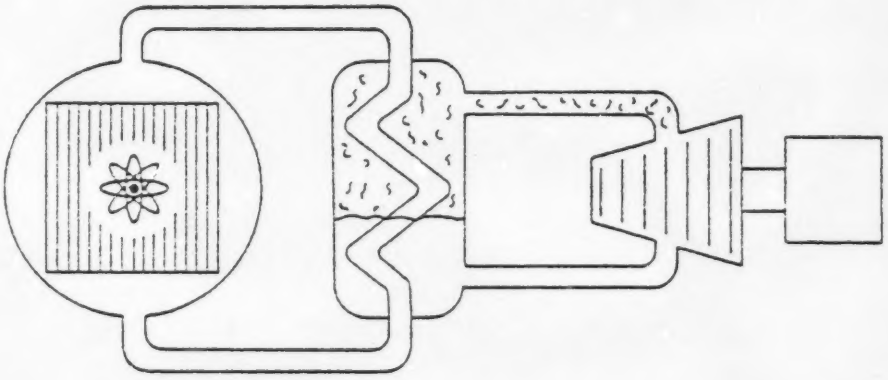
3. List materials used for BWR primary system liquid piping and explain how they are different for PWR primary system piping:

4. List materials and basic requirements for primary pipe insulation:

5. Define isolation valve:

NUCLEAR TECHNOLOGY

MODULE 5-5 NUCLEAR INSTRUMENTATION



INTRODUCTION TO NUCLEAR SYSTEMS

COURSE 5

TERC-SW

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST
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COURSE 5 INTRODUCTION TO NUCLEAR SYSTEMS

MODULE 5-1 Basic Nuclear Reactors Components
MODULE 5-2 Pressurized Water Reactors (PWR)
MODULE 5-3 Boiling Water Reactors (BWR)
MODULE 5-4 Primary Cooling Systems
MODULE 5-5 Nuclear Instrumentation
MODULE 5-6 Nuclear Fuel Cycle

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INTRODUCTION

The purpose of nuclear instrumentation systems is detection and measurement of nuclear radiation - all indications, recordings and controls point to protection personnel and equipment. In nuclear power plant applications, alpha particles, beta particles, gamma rays and neutrons are the significant types of radiation (from a detection and measurement point of view).

Devices for sensing the presence of nuclear radiation are called detectors. Radiation may be detected directly (as in the case of charged particles) or indirectly (as in the case of gamma rays and neutrons).

MODULE PREREQUISITES

Student should have completed all previous modules in this Course, as well as their prerequisites.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. Sketch and explain a basic, gas-filled radiation detector.
2. Describe how neutral particles, such as neutrons, produce ionizations in a gas-filled detector.
3. List four types of nuclear instrumentation systems found in a nuclear power plant.
4. List three major ranges for reactor power instrumentation; include type of detector used in each.
5. Describe applications of nuclear instrumentation in a reactor protection system.
6. List typical process streams (in a nuclear power plant) that are monitored for radiation level.
7. Measure neutron flux of a neutron source.

REVIEW

Most common radiation detectors employ gas-filled chambers in which radiation, passing through, interacts with the gas to form ion pairs. These ion pairs react to an electric charge on the chamber and cause current flow in the attached circuitry. The amount of current flow is proportional to the amount of radiation passing through the chamber. An axial center wire (Figure 5-5-1), electrically insulated from the cylindrical casing, is maintained at a positive potential relative to the casing by an external, variable, high-voltage power supply.

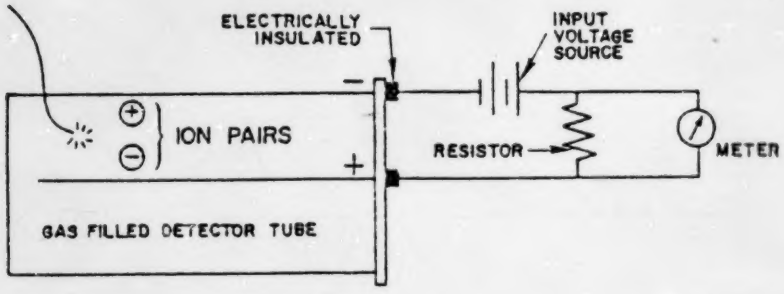


Figure 5-5-1. Gas-filled chamber.

When nuclear particles (or photons) pass through the detector, ion-pairs form in the gas along the path of the particle. Positive and negative ions, formed within the chamber, travel toward the negatively charged outer casing and positively charged center wire. Negative ions (electrons)

are attracted to and flow through the center wire and external circuit to the outer casing. There they replinish electrons (drawn from the outer casing to neutralize the positive ions) accumulated there. Electron flow in the external circuit is measured, indicating presence and quantity of radiation.

When a particle or ray (photon) of radiation enters an ionization chamber, the gas becomes a conductor because of the many ion pairs formed. Movement of ion pairs (toward center wire and can surface) makes the chamber behave as a closed switch, producing a current in the external circuit.

When all molecules have returned to their uncharged state and the gas becomes nonconducting, the chamber again acts as an open switch until the next radiation comes through.

While the radiation induced pulse is critical to the operation of a gas-filled detector, the resultant voltage across resistor R (Figure 5-5-2) is important.

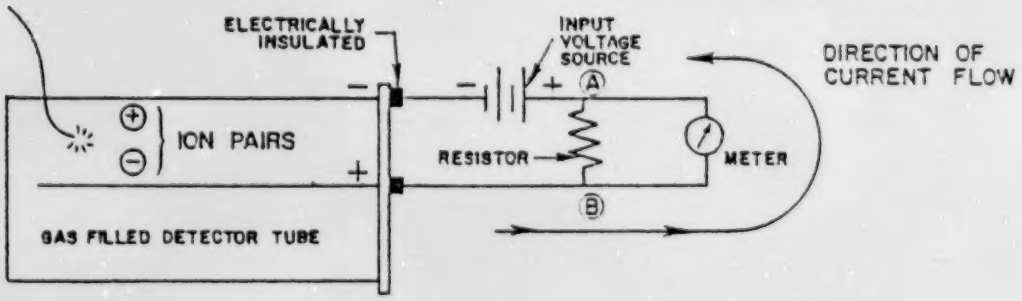


Figure 5-5-2. Ionization chamber conduction.

Ionization in the changer closes the circuit. Electrons tend to move through R, and encountered resistance causes a "pileup" at B. At A, there is a deficiency of electrons because electrons are leaving the chamber surface to neutralize the incoming positive ions. Thus, a potential difference is developed across R. The voltage across R is produced when radiation passing through the chamber closes the circuit and allows a current to flow. Thus, the signal that radiation has caused ionization in the chamber is the pulse of voltage across R, indicated by movement of the meter.

There are several types of ionization chambers. These vary in size, shape, materials and type of gas used. All ionization chambers detect radiation, but not all detect all types of radiation: materials and design determine type of radiation detected.

INSTRUMENTATION AND CONTROL SYSTEM APPLICATIONS

Nuclear instrumentation and control applications in a nuclear power plant can generally be divided into four types of systems:

Neutron monitoring.
Reactor protection.
Process radiation monitoring.
Area and environmental monitoring.

The first three systems are discussed below. (Area and environmental monitoring is described in another course.)

NEUTRON MONITORING SYSTEMS

Instrument systems must be provided to continuously measure reactor power level and rates of changes. This information, required by the operator to intelligently operate the reactor, also must be fed into the reactor protection system to shut the reactor down if certain limits are exceeded.

Conventional methods of measuring thermal power output (flow, temperature and pressure measuring instrumentation) do not respond quickly enough to protect against rapid power increases. Also, none of these parameters is necessarily proportional to reactor power. Neutron population, however, is proportional and can be quickly and accurately measured.

In general, a neutron monitoring system consists of a detector (usually a gas-filled ion chamber or proportional counter), plus an appropriate combination of electronic equipment, including high-voltage power supplies, amplifiers and readout devices.

A detector for a neutron monitoring system may be situated outside a reactor vessel or within the reactor core. If possible to do so without unduly sacrificing accuracy, the former arrangement is preferred because detectors tend to be fragile. Also, the environment outside a vessel is considerably less hostile than that within the core. (Temperatures are much lower, and the detector is subjected to less severe bombardment by ionizing radiation, which tends to result in gradual deterioration of insulation. If the detector is on the outside, it generally is in a holder near the external wall of the pressure vessel, at core elevation. Situated in this manner, the detector responds to neutrons that reach it after having leaked from the core, passed through the reflector and penetrated the wall of the vessel. If, on the other hand, an internal detector is used, neutron population is

is measured at various points in the reactor fuel region.

Source Range Monitors

The source range, as the name implies, is the lowest range in which the neutron level is measured. For this use, a detection device must be extremely sensitive. The detector for this system is usually a proportional counter, which combines desirable features of high sensitivity and fast response.

A proportional counter detects neutrons as follows: the detector has boron-10 (^{10}B) added to it, which, in the presence of thermal neutrons, undergoes the reaction $^{10}_{5}B$ (n,α) $^{7}_{3}Li$; ionization within the detector is produced by the alpha (α) particle and the ^{7}Li; the ^{7}Li ultimately becomes a neutral atom (as does the α, for that matter) but immediately after the reaction, it has considerable kinetic energy and "runs away" from its electrons for a moment, during which time it ionizes.

Since the range of an α particle or a ^{7}Li ion is very short, the boron must be located within the chamber itself if the particles are to reach the fill-gas. One way to accomplish this is to fill the detector with a gaseous compound of boron. The most common choice is boron trifluoride (BF_3), and such detectors are known as "BF_3 proportional counters." Another option is to coat interior surfaces of a detector can with ^{10}B. Such detectors, known as boron-lined proportional counters, or ^{10}B proportional counters, are filled predominantly with argon.

Another effective detector of neutrons is the fission chamber, an ion chamber with a coating of ^{235}U on the inside surface of the detector wall. Thermal neutrons fission this ^{235}U, and the highly charged fission fragments produce ioniza-

tion in the fill-gas.

The arrangement of electronic equipment in the startup channel varies somewhat from plant to plant. In a basic arrangement (Figure 5-5-3), however, a signal from the detector, which is very weak, is sent into the amplifier (AMP) to boost the signal power sufficiently to enable efficient monitoring. After amplification, the signal is sent to the signal-conditioning system, which performs a multitude of functions to produce the most accurate indication of neutron population. (Signal-conditioning is described in Module 6-3, "Neutron Flux Signal Conditioning.")

Pulses are sent to a device that extracts the logarithm of the count rate, and this value is displayed on a meter with a logarithmic scale, which generally reads out in "counts per second." The final signal is also sent to a device that computes its rate of change and determines the period (or startup rate). This value is also displayed.

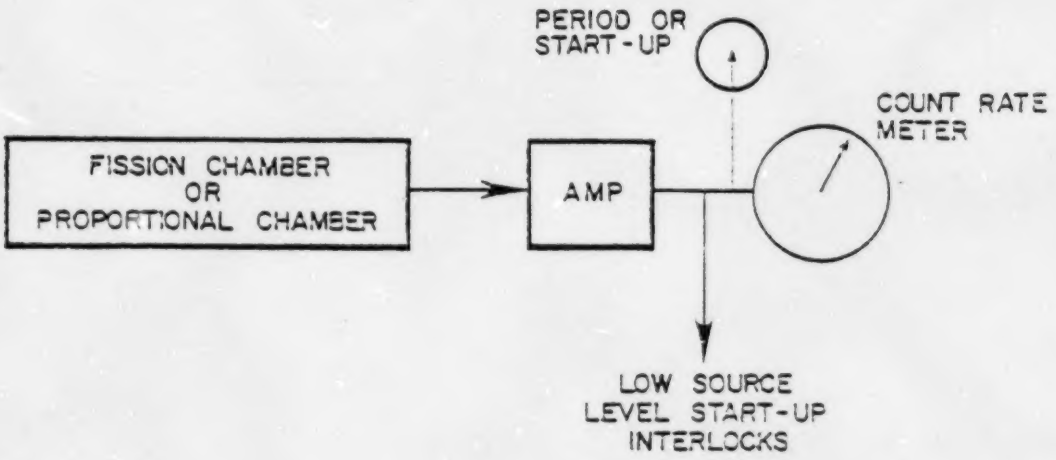


Figure 5-5-3. Typical source range monitoring system.

Finally, most reactors employ at least two independent source range monitoring systems. The detectors are located on opposite sides of the core so that at least one monitor is likely to see a rod withdrawal (Figure 5-5-4).

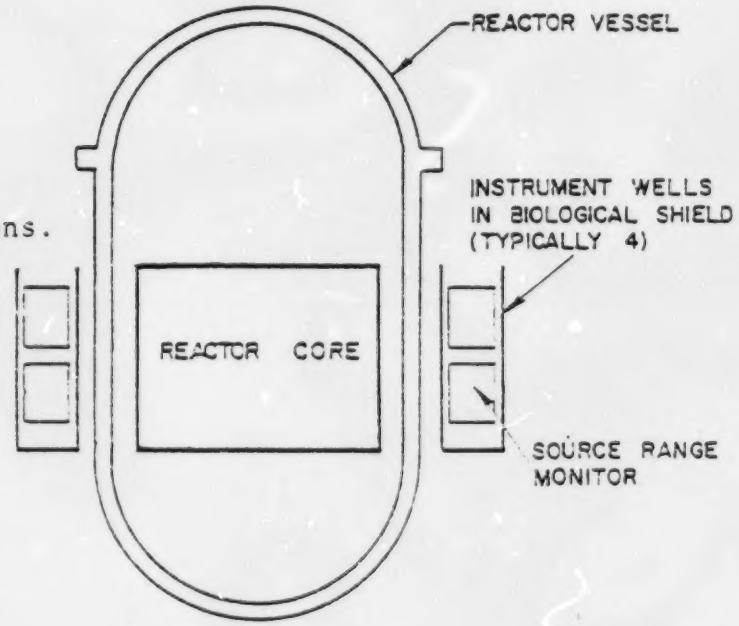


Figure 5-5-4.
Source range positions.

Intermediate Range Monitors

In the intermediate range, a neutron signal is fairly high; the most common systems employ ion chambers operating in the current mode. Since a neutron signal is still relatively low at the bottom end of the intermediate range, a means must be provided for distinguishing it from a γ signal. In these instruments, the current produced by γ interactions simply adds to that produced by neutron interactions, and it is impossible to tell what fraction of the current came from each source.

Therefore, a slightly different type of detector, called a gamma compensated ion chamber (Figure 5-5-5) is normally used in intermediate range monitoring systems that employ out-of-core detectors. A compensated ion chamber (CIC) actually

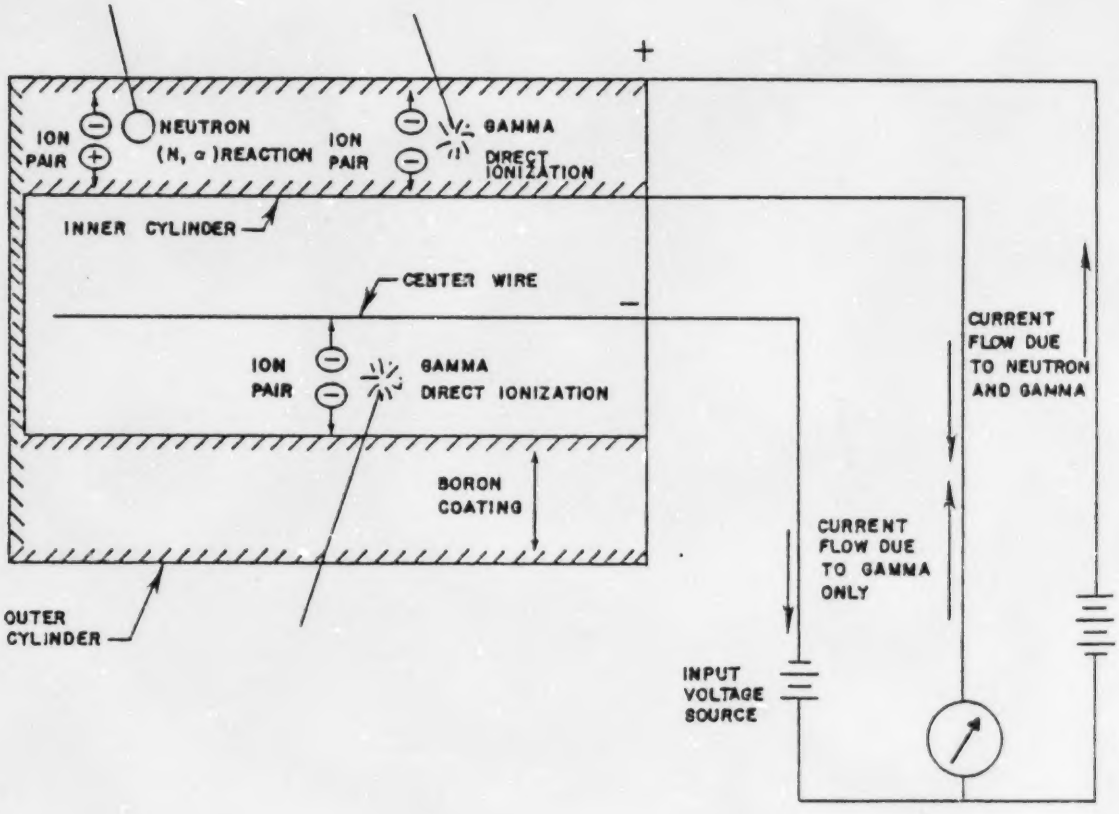


Figure 5-5-5. Simplified measurement circuit employing a compensated ion chamber.

is two ion chambers in a single case. One, coated with boron enriched in ^{10}B isotope, is sensitive to neutrons and gamma radiation. The other, uncoated, is sensitive only to gamma. By connecting the two so that their output currents electrically oppose one another, net electrical output, or current,

is the algebraic sum of the two currents.

If a neutron detector is perfectly gamma compensated, its output is proportional to the neutron flux at the detector location. Compensation is usually accomplished by varying the high voltage supplied to each chamber.

An intermediate range neutron monitoring system (Figure 5-5-6) is ordinarily provided with a logarithmic readout since it must indicate over several decades of reactor power. As in the source range, the rate of signal change is monitored and used for period and startup rate indication and protection.

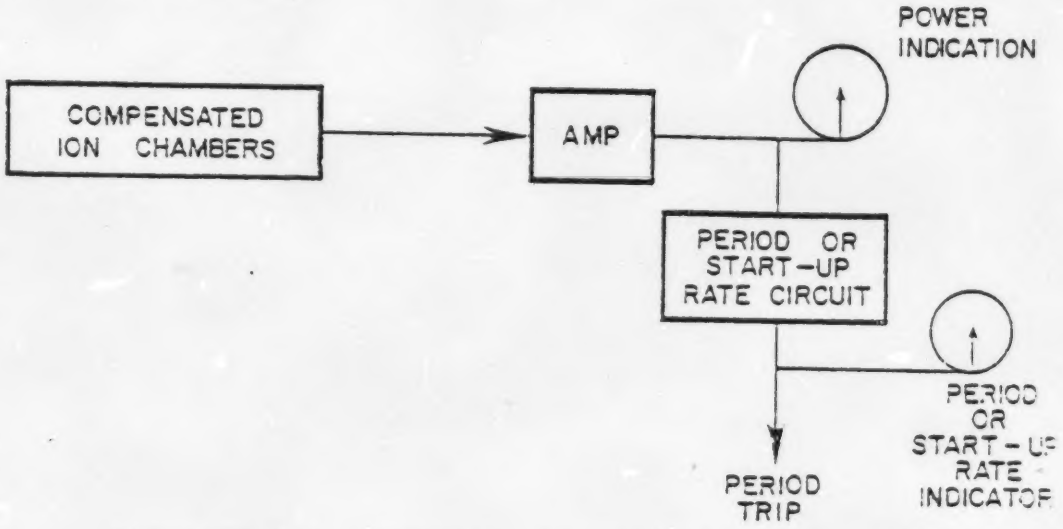


Figure 5-5-6. Typical intermediate range monitoring system, employing a CIC.

Most reactors use at least two intermediate range monitoring systems (with detectors appropriately located, whether in-core or out-of-core), to look at different core regions.

Power-Range Monitors

Power-range instruments are ordinarily ion chambers connected to linear readouts.

Large PWR's employ uncompensated ^{10}B-lined ion chambers in their power range systems. Compensation is not required because these instruments are not intended to respond until the neutron flux is high enough that the effect of γ radiation on indicated power level is minimal. Usually, four monitoring systems are used, with detectors situated in instrument wells (in biological shielding surrounding the vessel adjacent to each of the four "corners" of the core).

Large BWR's use miniature in-core fission chambers, both for local monitoring and for determining total core power. Some of the detectors are at fixed locations and others can be moved about.

In both PWR's and BWR's, automatic "scram" is associated with power level above a predetermined setpoint.

REACTOR PROTECTION SYSTEM

A reactor protection system includes safety sensors, which monitor critical plant parameters and equipment that uses the sensor output to shut the reactor down (or take some specified protection action) automatically. Protection functions include:

Scramming reactor.
Blocking control-rod withdrawals.
Actuating engineered safeguards, such as containment isolation and emergency core flooding and cooling.

One or more of these safety functions may be actuated

when a condition approaching unsafe operation is sensed. The action depends upon the nature and severity of the abnormal condition.

In one of the most popular types of safety system designs, the dual channel safety system, each safety channel is fed by at least two independent sensors from each measured variable (Figure 5-5-7). Operation of one sensor in a safety channel results in annunciation and a trip signal

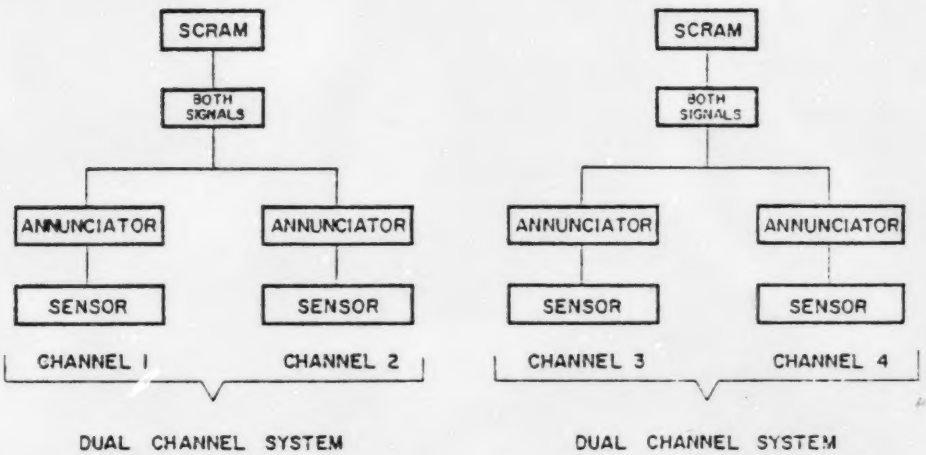


Figure 5-5-7. Typical dual channel safety system.

from that channel (often referred to as half-scram).

A simultaneous operation of a sensor in each of the two safety channels results in a reactor scram. Failure of one sensor to operate does not impair ability of the other sensors to operate. Also, false operation of a sensor cannot cause a reactor scram, since it trips only one safety chan-

nel. With this arrangement, any sensor can be calibrated, maintained or tested without causing a scram or without deactivating its trip. (Each safety channel receives its supply power from a separate source.)

Typical plant parameters connected to a reactor protection system include:

High reactor pressure.
Low reactor water level.
High neutron flux (high power).
Short reactor period.
High containment system pressure.
Low condenser vacuum.

Other parameters may be tied into the protection system, depending on reactor design, primary system and containment. Note: Manual reactor scram is always provided.

PROCESS RADIATION MONITORING SYSTEM

A process radiation monitoring system is provided to monitor and record radioactivity in process streams. It also annunciates any significant level increases. With few exceptions, these systems provide no automatic operation; rather, they provide the operator with information needed to made prudent decisions on plant operation.

In general, a process monitor consists of a detector connected to a logarithmic, remote readout in the control room. Process monitors mostly monitor the presence of fission products (and corrosion products, which are β-γ emitters). Since a detector ordinarily is situated outside of a pipe, or in a well within a pipe, it is apt to see more γ than β. Geometrical considerations may be more favorable for β de-

tection, however. In any event, the most commonly used detectors for these instruments are NaI scintillation detectors and GM tubes (because of their sensitivity). If the activity level in a process stream is expected to be high, an ion chamber may be satisfactory.

Typically monitored process streams include:

Stack or plant vent, where gaseous radioactive materials may be released.
Liquid waste discharge pipe.
Component (or closed) cooling water system.
Condenser air-ejector.
Reactor water.
Condensate or steam generator liquid.

In most instances, the purpose of these instruments is simply to inform the operators of the radiation status of the processes. In a few instances they perform automatic safeguard functions. In particular, the monitors on waste discharge lines operate automatic isolation valves to block release when waste radiation exceeds predetermined limits.

EXERCISES

1. Discuss principle of operation of ion chambers, proportional counters and GM tubes; include relative sensitivities, counting speeds, applicability to pulse and mean level usage and other advantages and disadvantages of each.

2. What is gas amplification? Is it strongly dependent upon applied voltage (to detector)?
3. Discuss principle of operation of a CIC.
4. Trace obtained on a startup channel employing a ^{10}B proportional counter and a logarithmic ratemeter with a maximum countrate capability of 10^5 is shown in Figure 5-5-8; discuss reason for behavior shown at points A and B.

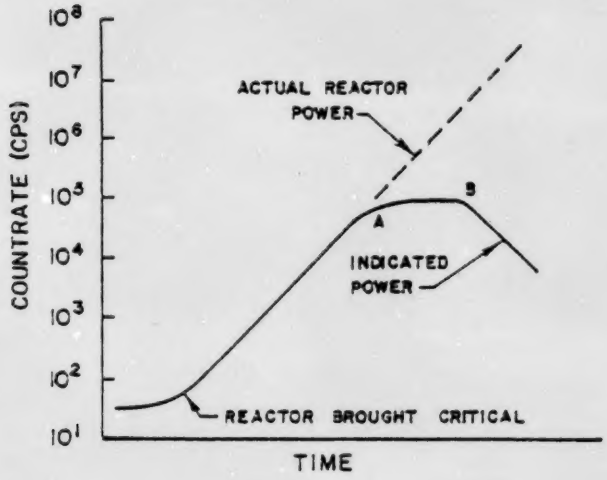


Figure 5-5-8.
Power level recorder.

MATERIALS

None.

PROCEDURES

None.

DATA TABLE

None.

REFERENCE MATERIALS

LaMarsh, John R. Introduction to Nuclear Engineering. Reading, MA: Addison-Wesley Publishing Co., 1975.

Lish, Kenneth C. Nuclear Power Plant Systems and Equipment. New York: Industrial Press, 1972.

1. Sketch and briefly explain a basic, gas-filled radiation detector:

2. Briefly describe how neutral particles, such as neutrons, produce ionizations in a gas-filled detector:

3. List four types of nuclear instrumentation systems found in a nuclear power plant:

4. List three major ranges for reactor power instrumentation, and type of detector used in each:

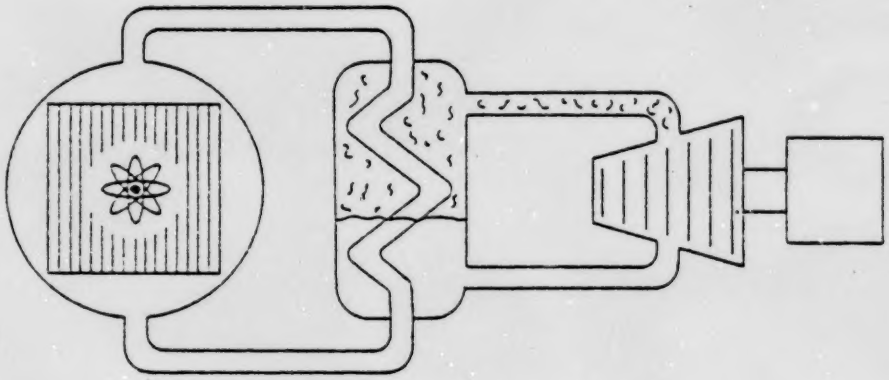
5. Briefly describe applications of nuclear instrumentation in reactor protection systems:

6. List six typical process streams (in a nuclear power plant) that are monitored for radiation levels:

NUCLEAR TECHNOLOGY

MODULE 5-6 NUCLEAR FUEL CYCLE. -

GEORGE J. STOLL, P.E.



INTRODUCTION TO NUCLEAR SYSTEMS

COURSE 5

TERC-SW 118

TECHNICAL EDUCATION RESEARCH CENTER — SOUTHWEST

4201 LAKE SHORE DRIVE WACO, TEXAS

COURSE MODULES

COURSE 5. INTRODUCTION TO NULEAR SYSTEMS

MODULE 5-1 Basic Nuclear Reactor Components

MODULE 5-2 Pressurized Water Reactors (PWR)

MODULE 5-3 Boiling Water Reactors (BWR)

MODULE 5-4 Primary Cooling Systems

MODULE 5-5 Nuclear Instrumentation

MODULE 5-6 Nuclear Fuel Cycle

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INTRODUCTION

This module examines the nuclear fuel cycle, which includes both the production of uranium fuel and the reprocessing of spent fuel (Figure 5-6-1).

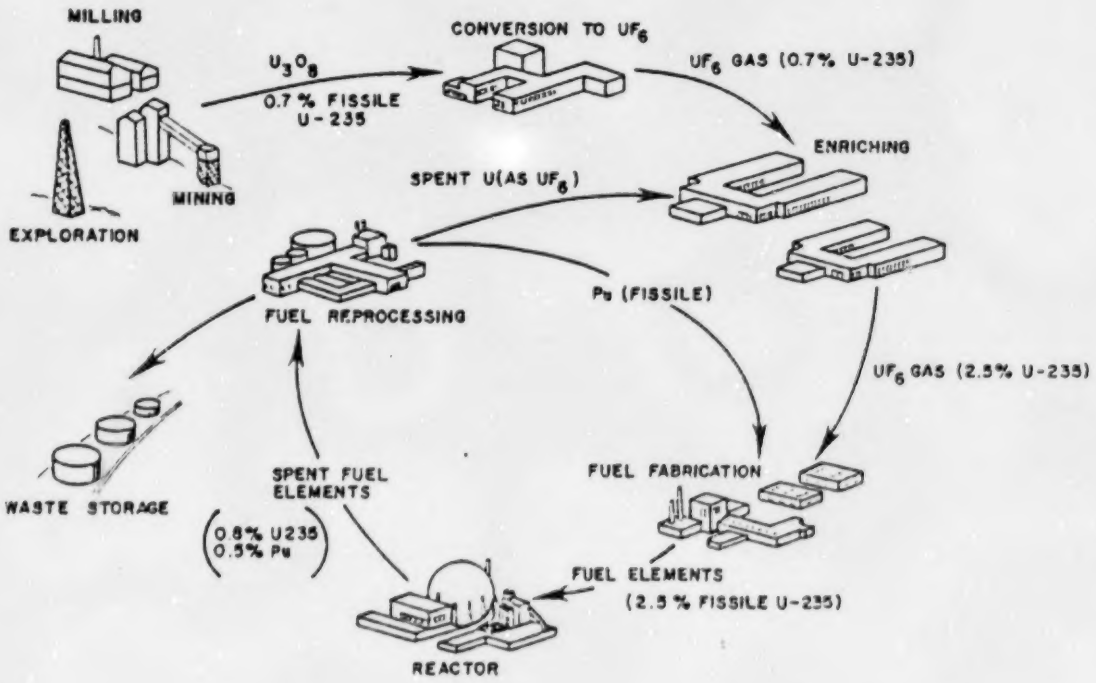

Figure 5-6-1. The nuclear fuel cycle.

MODULE PREREQUISITES

The student should have completed a radiation physics course and a course in basic chemical processes.

OBJECTIVES

Upon completion of this module, the student should be able to:

1. List major uranium sources in the United States.
2. List and discuss basic steps in "yellow cake" production.
3. List and discuss basic steps in uranium hexafluoride production; explain why uranium hexafluoride is important.
4. Explain "gaseous diffusion."
5. List the enrichment facilities (in U.S.).
6. Describe fabrication of uranium oxide fuel, fuel pellets and rods.
7. Sketch a flow chart; show major steps in fuel reprocessing.
8. Sketch a flow chart; explain total cycle, from mine to reprocessing, of a typical fuel element.

NUCLEAR FUEL

The basic raw material of the nuclear industry, uranium, is the heaviest natural element occurring in appreciable abundance. Its atomic number is 92, and it has a melting point of 2070°F. Also, it is slightly radioactive.

Natural uranium consists of three principal isotopes:

Uranium-234 (^{234}U) (0.01%).
Uranium-235 (^{235}U) (0.71%).
Uranium-238 (^{238}U) (99.28%).

^{235}U is a fissionable isotope, and ^{238}U is a fertile isotope, capable of being converted into fissionable material. All three emit alpha particles; however, only little care is needed in the handling of uranium in its natural state.

^{235}U, which is fissionable upon neutron absorption, is capable of sustaining a chain reaction. Its sister isotope, ^{238}U, is not capable of sustaining a chain reaction, although it can be fissioned by high-energy neutrons to some degree. ^{238}U, however, can capture neutrons and, ultimately, transforms into fissionable plutonium-239 (^{239}Pu), which is capable of sustaining a chain reaction. Because of this, ^{238}U is said to be a fertile material.

Because ^{235}U is the only fissionable material economically available in nature, and ^{238}U is a breeder of fissionable plutonium, they have become the basic raw materials for the nuclear power industry.

URANIUM MINING

Uranium is, by no means, a rare substance. It is estimated to constitute about 4 ppm of the earth's crust. Thus, uranium is more abundant than certain relatively familiar elements, such as silver and mercury.

Those primary minerals containing uranium include uranitite and pitch blend, both oxides of variable composition (ranging between UO_2 and U_3O_8), and carnotite ($K_2O.UO_3.V_2O_5.xH_2O$).

Major sources of commercial grade uranium ore, in the United States, are in the western states, especially the Colorado Plateau (Colorado, Utah, Arizona, Wyoming and New Mexico) where about 4 to 7 pounds of U_3O_8 can be extracted per ton of ore. Most production is from mines, although some open pit dragging is used.

MILLING

Mills extract uranium from the ore. These mills use one of several alternate processes tailored for the different chemical compositions of various ores. Basic steps of the most commonly used "acid process" include:

Step 1. Ore is mined, prepared and initially concentrated by grinding, crushing, washing and gravity separation.

Step 2. It is then digested in acid to leach out soluble uranium and other metals, such as vanadium, iron and aluminum.

Step 3. The ensuing liquor is then filtered and decanted in a series of steps to precipitate soluble metal impurities.

Step 4. Filtrate from the previous steps is treated with ammonia to precipitate $(NH_4)_2U_2O_7$.

Step 5. Precipitation is then converted to what is known in the industry as "yellow cake" - a crude uranium concentrate containing between 70% and 90% U_3O_8.

REFINING AND CONVERSION TO URANIUM HEXAFLUORIDE (UF_6)

Because uranium fuel must be essentially free from elements that have the ability to absorb neutrons needed for the fission process, yellow cake must undergo further purification. The yellow cake is converted into uranium hexafluoride (UF_6) gas, the compound most used in the enrichment process (Figure 5-6-2).

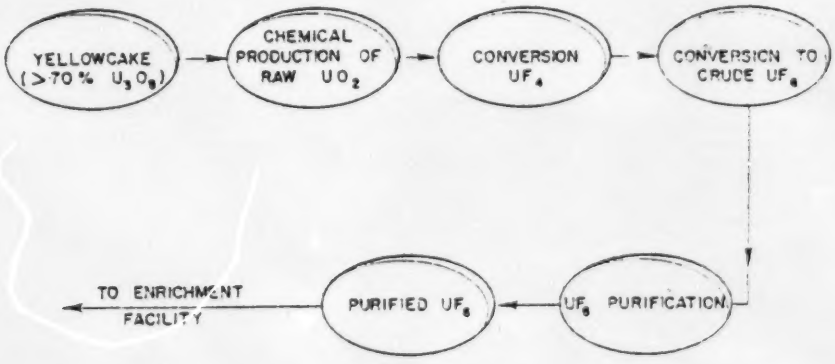

Figure 5-6-2. Process chart for UF_6 production.

The UF_6 now meets Federal Government specifications for impurity level and is ready to be sent to separation plants for isotope enrichment.

ISOTOPIC ENRICHMENT

All light-water moderated nuclear plants require enriched fuel (i.e., fuel with more than a normal amount of ^{235}U). Because uranium isotopes cannot be separated chemically, a process called "gaseous diffusion" is most frequently employed. This process is based upon the constant molecular motion of gases, with the lighter molecules possessing the highest velocities.

Consider a chamber separated into two compartments: a high-pressure compartment, containing a gas mixture, and an evacuated compartment (Figure 5-6-3). The separating medium is a thin, porous material containing hundreds of millions of very small openings per square inch. The aver-

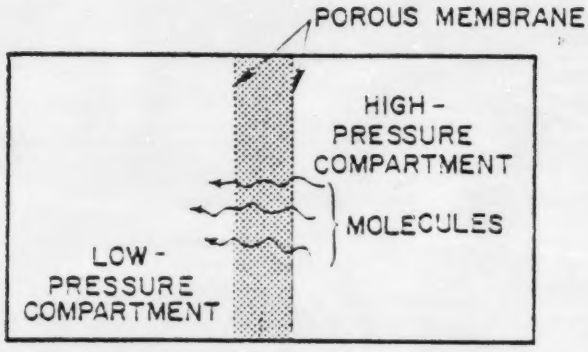

Figure 5-6-3.
Gaseous diffusion process.

age diameter of these pores is about two millionths of an inch.

The gas gradually diffuses through the barrier (due to the pressure differential across it); however, the light molecules diffuse faster because of their higher velocity. They strike the barrier more frequently and have a better chance of finding an opening; therefore, the gas that passes through the barrier first is richer in the lighter isotope (^{235}U), and that remaining behind has a higher proportion of the heavier isotope (^{238}U).

A mixture of ^{238}U hexafluoride and ^{235}U hexafluoride behaves similarly. The degree of separation is very small, and the process must be repeated many times before practical enrichment is possible; therefore, a gaseous diffusion plant must consist of hundreds of similar stages connected in series, with the arrangement of these stages called "diffusion cascade."

In each stage, a pump compresses the gas to the next stage pressure and a cooler removes the generated heat of compression. The gas becomes more enriched as it travels up the cascade. The degree of enrichment depends on the number of stages in the overall system.

Depleted uranium (known as "tails") is withdrawn at the bottom of the cascade. Gaseous diffusion plants, under Federal direction, are situated at Oak Ridge (Tennessee), Paducah (Kentucky), and Portsmouth (Ohio).

OXIDE FUEL FABRICATION

All present day BWR's and PWR's use slightly enriched UO_2 fuel. The first step in the manufacture of fuel assemblies is conversion of UF_6 gas (of the proper enrichment) into UO_2. (This conversion, a complex chemical process, is not discussed here.) The UO_2 produced is compacted and sintered into cylindrical pellets with about 95% of the theoretical maximum density for UO_2, which allows escape paths for the fission product gases from the pellet. (Each pellet is about 1/2 inch in diameter and 1 inch long.)

FUEL ASSEMBLY FABRICATION

The UO_2 pellets are arranged into stacks (at the fuel-assembly fabrication plant) and checked for uranium content. They are then weighed and inserted into Zircaloy-2 tubes (Figure 5-6-4). These tubes have been provided with springs to hold the pellets in place. A plenum, containing helium, accomodates buildup of fission product gases throughout the active lifetime of the rod. (Helium is used because it is inert and does not react with rod components; also, it has moderator properties similar to those of water.) The tube ends are seal-welded shut.

Tubes are assembled into a desired array, with instrument tubes and control-rod guide tubes placed in appropriate locations. The assembly (Figure 5-6-5) then receives end fittings for handling and placement into core. Quality-

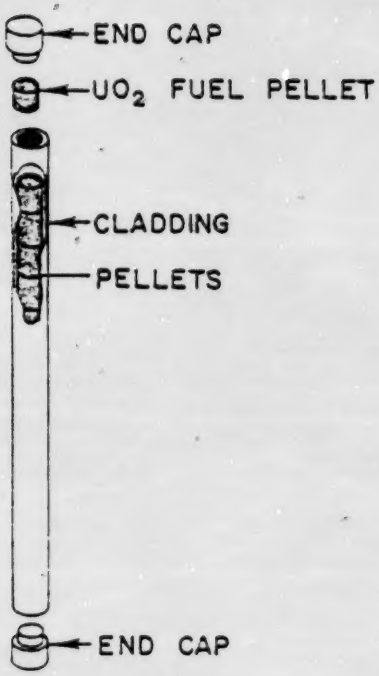

Figure 5-6-4.
PWR UO_2 fuel rod.

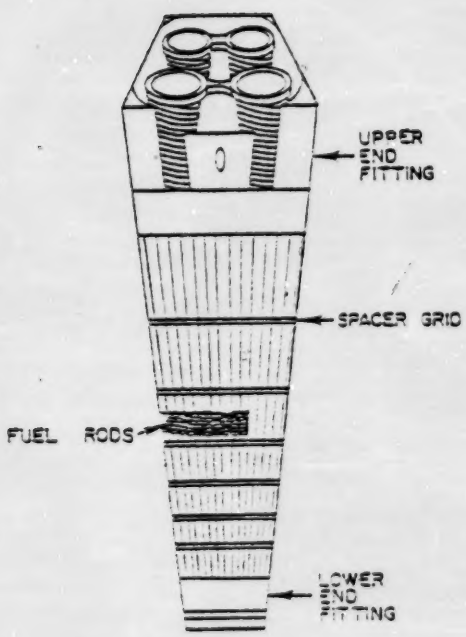

Figure 5-6-5.
Typical fuel assembly.

control checks are extremely rigid during this entire fabrication process.

SPENT-FUEL SHIPPING

After completing an assigned period of time in a reactor, and a long period in a spent-fuel pit (while its high radiation level decays), the spent fuel is shipped to a fuel reprocessing plant. There, residual uranium and plutonium, which has been bred (generated) are reclaimed.

REPROCESSING SPENT FUEL

At the end of its useful life, spent fuel arrives at a reprocessing plant. There, unused ^{235}U and ^{239}Pu are recovered. (At the present time, no commercial fuel is reprocessed because of Government policy. The following discussion assumes that these Government controls are relaxed and that reprocessing is allowed.)

The solvent extraction method (Figure 5-6-6) is one procedure used to reprocess. Its basic steps include:

Cladding removal (mechanically or chemically) - e.g., Zircaloy is dissolved in hydrofluoric acid.

UO_2 is dissolved in nitric acid.

Uranium, plutonium and fission products are separated by solvent extraction processes. (Solvent extrac-

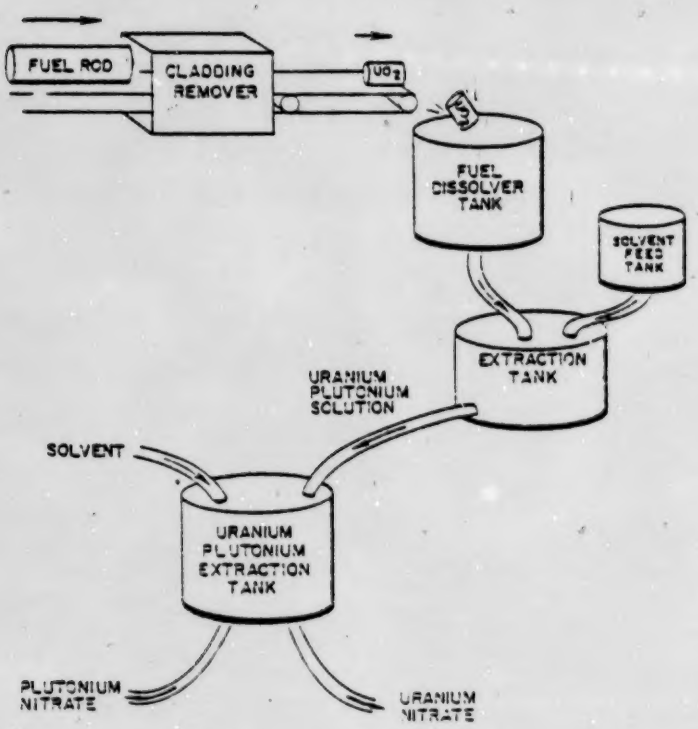

Figure 5-6-6. Reprocessing spent-fuel assemblies.

tion works on the principle that uranium is more soluble in certain organic liquids than in water; therefore, if the proper organic solvent is added to an aqueous solution of uranium salt, the uranium enters the organic solvent. The solvent chosen must not mix with water: if so, the two separate, and the uranium is separated from the more soluble solution.)

Uranium and plutonium are purified by additional solvent extraction, and then concentrated.

Potentially valuable isotopes, such as neptunium-237 (^{237}Np), are contained in liquid waste from repro-

cessing. (Note: No attempt is being made presently to separate these constituents because the demand is not great enough to make such steps economical. The waste products are concentrated, neutralized and stored in underground liquid waste tanks. These stored fission products remain available if separation becomes economically feasible.)

Part of the uranium recovered from the reprocessing cycle can be used again as fuel by re-enrichment in a gaseous diffusion plant. Uranyl nitrate recovered must first be reconverted to UF_6, which can be accomplished in a manner similar to that used on natural uranium.

PLUTONIUM RECYCLE

Because potential reserves of ^{235}U are limited, it is considered by many that plutonium must provide the basic fuel for the next generation of power reactors (the breeders). Certain technical difficulties have prevented its use. These problems involve its extreme toxicity and physics problems (reduced delayed neutron fractions).

Most of the world's plutonium, prior to the 1960's, came about as part of military use; however, plutonium is also generated in power reactors and its recovery is significant for the nuclear industry because it can subsequently be used as nuclear fuel, and recycled back into a light-water reactor (known as the "plutonium recycle").

Plutonium does not require expensive isotopic enrich-

ment, as does uranium. It is a natural fuel for fast breeder reactors. Chemical properties associated with it increase the difficulty of fabrication of fuel elements, however, making the overall cost of plutonium fuel greater.

THORIUM CYCLE

Another possible fuel material for nuclear power is thorium, in the form of thorium oxide (ThO_2). Essentially a fertile material, in a reactor, ThO_2 absorbs a neutron and becomes uranium-233 (^{233}U), a fissile isotope of uranium that does not occur in nature.

EXERCISES

1. Define:
   - Yellow cake.
   - Milling.
   - Isotopic enrichment.
   - Diffusion cascade.
   - Transfer cask.
   - Cladding.
   - Solvent extraction.
2. Sketch and label a nuclear fuel cycle.
3. Briefly describe isotopic enrichment by gaseous dif-

fusion.

4. Sketch a simple flow chart of uranium enriching cascade.
5. List three plants where enrichment is carried out.
6. List major U.S. areas that are natural uranium sources.
7. Briefly describe fuel assembly fabrication.
8. Briefly describe how UF_6 is converted to oxide fuel pellets.
9. Briefly describe liquid waste disposal procedures.
10. Briefly describe problems in using plutonium and thorium as nuclear fuel sources.

MATERIALS

None.

PROCEDURES

1. View film(s) of a uranium mining facility:
   Describe type of mine.
   Describe chemical analysis of ore.
2. View film(s) of a uranium milling facility:
   Describe type of milling process observed.
   Describe types of machinery observed.

3. View film(s) of an assembly fabrication plant:
   Describe and note material accounting system.
   Observe end-plug welding procedure.
   Observe and describe shipping procedures.

DATA TABLE

None.

REFERENCE MATERIALS

Lish, Kenneth C. Nuclear Power Plant Systems and Equipment. New York: Industrial Press, 1972.

McPhee, John A. The Curve of Binding Energy. New York: Farrar, Straus and Giroux, 1974.

U.S. Department of Energy. Understanding the Atom Series Booklets. Oak Ridge, TN: Technical Information Office.

1. List major sources of uranium used in U.S.:

2. List and briefly discuss basic steps in yellow cake production:

3. List basic steps in uranium hexafluoride production; explain why uranium hexafluoride is important:

4. Sketch and explain the "gaseous diffusion" process:

5. List present enrichment facilities in U.S.:

6. Briefly describe fabrication of uranium oxide fuel, fuel pellets and rods:

7. Draw a flow chart, showing major steps in fuel reprocessing:

8. Draw a flow chart showing a total nuclear fuel cycle, from mine to reprocessing:

NUCLEAR TECHNOLOGY

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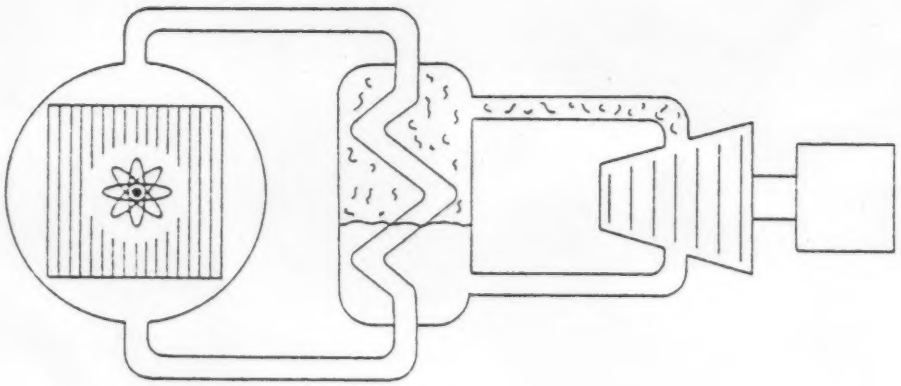
COURSE 5 - INTRODUCTION TO NUCLEAR SYSTEMS



INTRODUCTION TO NUCLEAR SYSTEMS

COURSE 5

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NUCLEAR TECHNOLOGY SERIES

INDEX

for

COURSE 5, INTRODUCTION TO NUCLEAR SYSTEMS (SIX MODULES)

NOTE: Each module in this Course is numbered separately, and the page numbers begin with page 1 in each module. Therefore, this Index lists for each entry the module number (underlined), followed by a comma and the page number. Figures and tables are listed by their numbers, followed by the module and page number in parentheses. A Master Index is available in a separate volume. This general index merges all index entries from the 35 Nuclear Courses. It is a valuable research tool, with which the student or researcher can locate all the references to any subject within the 35 Courses.

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